TECHNICAL COMMITTEE ON THERMAL REACTOR SAFETY RESEARCH

PROCEEDINGS OF THE SIXTH TECHNICAL COMMITTEE MEETING
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN
VIENNA, AUSTRIA, 8-11 JUNE 1987

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1987
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Chairman: D. Haschke
The Swiss Federal Institute for Reactor Research Würenlingen, Switzerland

Scientific Secretary: M.W. Jankowski
International Atomic Energy Agency Vienna, Austria

INTERNATIONAL ATOMIC ENERGY AGENCY VIENNA, 1987
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PARTICIPATION

Members of the TC - SR

Mr. J. Sainsbury          Canada
Mr. M. Hrehor             Czechoslovakia
Mr. L. Mattila           Finland
Mr. R. Zammite           France
Mr. M. Banaschik         Germany, Fed. Rep. of
Mr. L. Szabados          Hungary
Mr. L. Voross            Hungary
Mr. A. Anunziato          Italy
Mr. J. Sánchez Gutiérrez  Mexico
Mr. K. J. Brinkmann       Netherlands
Mr. E. Józefowicz        Poland
Mr. J. E. de Carlos       Spain
Mr. M. Montes             Spain
Mr. D. Rafael San Martin  Spain
Mr. L. Hammar            Sweden
Mr. D. Haschke            Switzerland
Mr. T. Türker             Turkey
Mr. V. G. Asmolov         USSR
Mr. J. I. Bramman         UK
Mr. P. G. Bonell          UK
Mr. J. L. N. Cortez       USA
Mr. B. Mavko              Yugoslavia
Mr. D. Feretic            Yugoslavia
Mr. M. A. Markovina       CEC
Mr. W. Haussermann        OECD
IAEA Staff Members

Prof. L. Konstantinov
Mr. M. Rosen
Mr. E. Yaremy
Mr. A. Karbassioun
Mr. V. Tolstykh
Mr. F. Amano
Mr. M. Jankowski
(Scientific Secretary)

Deputy Director General,
Dept. of Nuclear Energy and Safety
Director,
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Division of Nuclear Safety
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Division of Nuclear Safety
Division of Nuclear Safety
The Technical Committee on Thermal Reactor Safety Research (TC-SR) held its sixth meeting from 8-11 June 1987 at the Agency's Headquarters in Vienna. It was attended by 25 participants representing 18 countries and 2 international organizations.

The meeting was opened by Prof. L. Konstantinov, the IAEA Deputy Director General, Director of the Department of Nuclear Energy. In his opening remarks he stressed the need for continued co-operation and the exchange of safety related research results. This co-operation among all Member States, as well as increasingly effective co-operation between international organizations (the OECD/NEA, the CEC and the CMEA), has contributed in the past to significant improvements in nuclear power plant performance, reliability and safety.

Many of the research programmes undertaken or being planned in Member States will serve as bases for introducing further safety improvements and modifications to nuclear power plants operation and management. The increased attention to selected and important topics - such as analyses of accidents beyond the design basis, accident prevention management, and consequent mitigation - is already reflected in research activities currently being performed. The results are awaited with great interest. He also stressed the planned expansion of the IAEA Co-ordinated Research Programmes (CRP), not only enabling the greater exchange of information on specialized topics, but allowing the formation of complimentary research projects in certain Member States. The work and the recommendations of this Technical Committee on topics of mutual interest and recommended future activities will undoubtedly provide a significant input for the formulation of the research goals and objectives to be achieved through the CRPs.
1. With respect to exchange on national research activities the committee members presented their most recent activities in the area of nuclear safety research. The main areas of activity reported during national presentations are as follows:

**Summary of Topics of National Research Programmes**

**AUSTRIA**

(1) Thermal hydraulics and fuel behaviour
(2) Two phase flow instrumentation

**CANADA**

(1) Thermal hydraulics
    . RD-14 CANDU heat transport loop with steam generators, ECCS
    . Large Scale Header Test Facility
    . Cold Water Injection Test Facility
(2) Fuel element thermal and mechanical behaviour
(3) Fuel channel thermal and mechanical behaviour
(4) Fission product chemistry and transport
(5) Containment behaviour

**CZECHOSLOVAKIA**

(1) Transient analysis
(2) Loss-of-coolant accidents
(3) Fuel behaviour in accident conditions
(4) Dynamic response of structures
(5) PSA
FINLAND

(1) Severe accidents, source term and accident management
(2) Safety analysis tools
(3) Operator support systems
(4) New thermal hydraulic test facility
(5) Leak before break
(6) Ageing

FRANCE

(1) Human factor
   . operator training
   . man/machine interaction
   . safe operational organization
   . sound procedures
(2) Accident management
(3) Containment performance

FEDERAL REPUBLIC OF GERMANY

(1) Component safety and quality assurance
(2) Plant transients and accident sequences
(3) Man/machine interaction
(4) Risk and reliability

HUNGARY

(1) Plant transients and SBLOCA
(2) Severe accident analysis
(3) PSA
(4) Man/machine interaction
ITALY

(1) Structural integrity
(2) LOCA and transients
(3) Plant-environment interaction
(4) Severe accident analysis and risk evaluation

MEXICO

(1) LOCA analysis
(2) Specific studies related to station black-out
(3) Use of microcomputers in PSA
(4) Level 1 PSA for the Laguna Verde NPP
(5) Assessment of the source term for the Laguna Verde NPP
(6) Fuel management

SPAIN

(1) Prevention of accidents
   . stress crack corrosion in pipes and components
   . embrittlement of vessel material
   . non-destructive examination
(2) Loss of coolant and severe accidents
(3) PSA

SWEDEN

(1) Human factors
(2) Structural integrity and materials
(3) Thermalhydraulics
(4) Source term
(5) Severe accidents and plant internal accident management
(6) Systematic safety analysis
(7) Emergency and early warning monitoring system
(8) Nuclear waste
SWITZERLAND

(1) Operational safety and prevention of severe accidents
(2) Severe accidents and source term

TURKEY

(1) CRP on PSA for research reactors
(2) PS evaluation project (to assist the licensing of CANDU 600 NPP)
(3) Core conversion (to replace highly enriched fuel with low enriched)

USSR

(1) Hydrodynamic and heat exchange studies
   - large-scale experiments
   - effectiveness of ECCS
   - fuel behaviour
   - diagnostics
(2) Fuel elements and assemblies behaviour in accident conditions
(3) Fuel behaviour and source term during severe accidents
(4) Emergency planning
(5) Hydrogen
(6) Reliability analysis and PSA

UNITED KINGDOM

(1) Degraded core studies
(2) Fuel coolant interactions and fuel material studies
(3) Heat transfer and hydraulics
(4) Severe accidents
(5) PSA
USA

(1) Integrity, operability and ageing of structures and components
    • primary system integrity
    • ageing of components, systems and structures
    • external events
    • containment integrity

(2) NPP systems safety function reliability
    • plant performance
    • human performance
    • operational safety reliability
    • accident management

(3) Public protection from radiation
    • source term and containment loads
    • accident risk methods and analysis
    • severe accident regulatory implementation

(4) Waste management

YUGOSLAVIA

(1) PSA
(2) Deterministic safety analysis
(3) Structural analysis

CEC/Ispra

(1) Reliability and risk evaluation
(2) Project for inspection of steel components (PISC)
(3) LWR primary circuit components life prediction
(4) Study of abnormal behaviour of LWR cooling systems
(5) Source term
(6) Containment studies
OECD/NEA

(1) Operating experience and human factors
   . IRS
   . human factors

(2) Reactor transients and LOCAs
   . LOFT project
   . HALDEN project

(3) Primary circuit integrity
   . fracture mechanics
   . ageing
   . NDE

(4) Source term and accident consequences

(5) TMI-2 examination

(6) Severe accidents

(7) Containment

(8) Risk assessment

(8) Fuel cycle safety

The full text of the national presentations are provided in the Annex I.
2. The committee members were updated on the progress and recent results of the LWR Aerosol Containment Experiments (LACE) and the Advanced Containment Experiments (ACE) Programmes.

The LACE programme, sponsored by an international consortium, is investigating inherent aerosol behaviour for three postulated high consequence accident sequences: the containment by-pass or V-sequences, failure to isolate containment and delayed containment failure.

The ACE complements the LACE programme by investigating additional fission product generation and deposition phenomena involved in degraded reactor containment building conditions and means of controlling fission product release using passive filtering devices: the objectives of the ACE programme are: (1) to provide a comparative experimental basis for filtration techniques (e.g., submerged gravel beds, water pools, sand beds, etc.); (2) to provide data for modelling iodine species transport; (3) to determine fission product releases from molten corium-concrete reactions; and (4) to develop and validate the applicable computer codes and models.

In the absence of Mr. F. Rahn from the Electric Research Power Institute (EPRI), USA, the technical aspects were presented by Mr. D. Haschke, Switzerland. The programme received a considerable amount of interest, and the Agency encouraged further exploration of participation by any of the interested Member States on a bilateral basis.

In addition, the recommendation was formulated to the IAEA for further exploration and collecting of information on programme development.

3. The committee members were informed on the progress for the joint publication (IAEA/OECD-NEA/CEC) of the Nuclear Thermal Research Index (NTRI). The first joint publication is planned by the end of 1988; the second edition is planned for publication by the end of 1989. The first and second edition printed by the IAEA will be treated as trial exercises. After the second edition subsequent publication is recommended every two years.
4. The participation in the OECD/NEA International Standard Problem (ISP), utilizing the Italian SPES (Simulazione PWR per Esperienze di Sicurezza) facility, was opened to the non-OECD countries. The SPES facility is designed to simulate thermal-hydraulic phenomena of small break LOCA and operational transients. The main objective of the ISP is to benchmark the large system thermal-hydraulic computer codes and to quantify some of the uncertainties in code predictions.

Informally, the following countries expressed their interest in the ISP participation: Bulgaria, Czechoslovakia, China, Finland, Hungary, Poland and Yugoslavia. It was recommended that the invitation to participate in the ISP be forwarded to the non-OECD Member States by the IAEA.

5. The list of important topics for future activities was extensively discussed and reviewed. The following list of topics and their priorities was agreed on:

   (1) Containment behaviour and performance
       (a) hydrogen
       (b) fission product transport and behaviour
       (c) thermal hydraulics
       (d) containment behaviour and structural integrity
       (e) containment management

   (2) Operational safety
       (a) man-machine interface
       (b) human factors
       (c) early diagnosis of failures

   (3) Severe accidents and accident management
       (a) all types of nuclear power reactors
       (b) analyses of severe accident consequences
       (c) accident management
           (accident prevention and mitigation)
       (d) source term
       (e) cesium management
(4) Loss-of-coolant accidents (LOCAs) and reactivity initiated accidents (RIAs)

(5) Adequacies of emergency core cooling systems

(6) Seismic considerations in design

(7) Plant ageing phenomena and degradation in service

(8) Primary circuit integrity, including structural mechanics aspects, fracture mechanics and intergranular stress

(9) Corrosion cracking
Reliability methodology for application in PSA

(10) Safety impact of control and instrumentation systems.

6. The technical committee recommended the following topics for meeting to be considered in 1988/89 on the basis of a revised list of important topics:

(1) Containment performance and behaviour (1988/89)
(2) Early diagnosis of failures (1988)
(3) Loss-of-coolant accidents, special transients and adequacy of emergency core cooling (1989)

Regarding the meeting on early diagnosis of failures, Czechoslovakia informally expressed an interest in hosting this specialists meeting; Hungary expressed (also informally) interest in hosting the meeting on LOCA, special transients and adequacy of emergency core cooling.

7. The technical committee agreed that the next meeting will be held 7-10 June 1988 in Vienna, Austria.
ANNEX I

Updated Information on National Research Activities
THERMAL REACTOR SAFETY IN AUSTRIA

As now all political parties in Austria have committed themselves to prevent nuclear power to be used as an energy source in Austria the funding for reactor safety research becomes increasingly difficult. On the other hand we are surrounded by countries which operate NPPs. This makes it necessary to maintain at least a certain level of nuclear expertise. The Tschernobyl accident also showed the need of both government and public to have experts at hand. Nuclear expertise is impossible without international contacts, at least for a country like Austria. This is the reason why we are grateful for activities such as this committee. We try to support that by contributing as much as possible to the international reactor safety research.

Our programme is centered around the following topics:

Thermal hydraulics and fuel behaviour:

Within the OECD-LOFT Project one of our colleagues is assigned to EG&G, Idaho, USA. In the course of the analysis of OECD LOFT-FPI we are working on the burst data using our own code BALO-2A. We finished the IAEA Standard Problem FMK-NVH and are currently working on the OECD International Standard Problem PIPER (Pisa, Italy) using the Japanese Code THYDE-B. In collaboration with Israel we are investigating some thermal-hydraulic aspects of BWRs. As soon as funding is available we will begin to participate in the International Thermohydraulic Code Assessment and Application Program (ICAF) of the USNRC using RELAP 5 Mod 2.

Two phase flow instrumentation:

In collaboration with KFKI-Budapest we are developing, building, testing and selling instruments for measuring two phase flow. For the analysis connected with this work a collaboration with the People's Republic of China has begun.
This paper summarizes the continuing research and development programs which are being performed in support of CANDU reactor licensing. The purpose of the various programs is to advance the understanding of the basic phenomena and fundamental processes to develop improved predictive capabilities in the models which are used in the safety assessment of CANDU reactors. Through this ongoing process, the uncertainties in the safety analyses can be reduced. It also allows identification of unnecessary restrictions in design or operation and so lead to improved operating costs and efficiency.
1. INTRODUCTION

Research and development in support of CANDU-reactor licensing, can be divided into five fundamental areas. These are:

1) system and subchannel thermalhydraulics
2) fuel element thermal and mechanical behaviour
3) fuel channel thermal and mechanical behaviour
4) fission product chemistry, and transport; and
5) containment behaviour

The approach which has been taken in investigating these fields under the high temperature transient conditions that are expected under various postulated severe accident scenarios, involve the establishing of a sound theoretical basis for models, which are then verified systematically against experiments. Starting with the fundamental physical principles, models are developed for important processes and components. Predictions using these models are then compared against experiments in separate effects tests. Once these models are sufficiently verified, they are combined to form integrated models which are then checked against experiments that have the essential geometric and physical characteristics of the reactor system. The products are computer models that give predictions of known accuracy.

This paper provides a basic overview of the various research and development programs which are currently underway in support of CANDU reactor licensing. It intentionally does not provide technical details as these can be obtained from reading the numerous research reports, some of which are referenced in this paper. This report continues on from that reported in the previous year.
2. **THERMALHYDRAULICS**

The accident thermalhydraulics test program included experiments in 3 test loops: RD-14 (CANDU heat transport loop with steam generators, ECCS); Large Scale Header Test Facility (LSHTF) (full-size reactor supply headers feeding 30 individual channel representations); Cold Water Injection Test Loop (CWIT), (detailed representation of single CANDU fuel channel and feeders).

2.1 **RD-14**

RD-14 is an 11 MW full height, full length representation of a CANDU primary heat transport system. In its present configuration it contains two full-scale heated channels. The facility is used to verify assumptions made in models to predict reactor behaviour under postulated accident conditions.

Over the past year, an extensive experimental survey has been carried out investigating loop behaviour for: large LOCA in various PHTS pipes; small LOCA; thermosyphoning with varying inventory at high and low pressure, high and low power. Also, the loop has been modified to make the steam generators more proto-typical in recirculation flow, response to power reduction.

The present RD-14 program is almost complete. The loop will be shut down for the 2nd half of 1987 for modifications into a multiple-parallel-channel facility (5 fuel channels per core pass). This facility will be commissioned in early 1988, and will allow investigations into parallel channel effects during LOCA blowdown and refill, and during low inventory thermosyphoning.

2.2 **Large Scale Header Test Facility**

This facility was used to study flow distribution between fuel channels under accident conditions, (a) steady-state two-phase coolant flow through the header, (b) refill of a steam-filled header and fuel channels by ECCS. Tests were carried out in each case for a range of flow rates and pressures, to enable mapping of behaviour against these parameters. The test results will improve understanding of the phenomena involved in header flow distribution.

2.3 **Cold Water Injection Test Facility**

During 1986-87, a series of ECCS refill tests was conducted with this rig using different feeder pipe diameters, to study the affect of pipe diameter on feeder refill rate. Tests included studies with different feeder orifices and with non-condensible gas injection.
3. FUEL ELEMENT THERMAL AND MECHANICAL BEHAVIOUR

Construction of the Blowdown Test Facility (BTF) at CRNL is basically complete. The first series of blowdown tests will start in 1987. The first test will be performed with a trefoil of zircaloy sheathed CANDU fuel elements (one fresh and two irradiated). Conditions are such that temperatures of up to 1500°C should be attained. The purpose of the tests is to:

1) measure active fission product release from the fuel during a high temperature transient in the temperature regime where releases are rapid.
2) measure the additional release and assess the fuel damage due to a water quench of the fuel element.
3) measure the transport and deposition of released fission products, and identify the form in which they move, and their partitioning.

Experiments were also performed to investigate the heatup characteristics of a CANDU fuel bundle in air, to understand the behaviour following various postulated fuel transfer accidents. The different environmental conditions investigated were; heatup in both stagnant and turbulent air and fog, as well as in light and heavy rain conditions.

Oxidation of UO₂ in air or steam can cause enhanced fission product release, by mechanisms such as increased diffusion, grain growth, or volatilisation of higher oxides of uranium. As much as 100% release of Cs or Iodine was observed after heating UO₂ in air at 1100°C for 30 minutes. Experimental and analytical work on the oxidation of UO₂ as functions of parameters such as time, temperature, oxygen partial pressure and surface morphology have been completed, and the understanding incorporated in predictive codes. The influence of oxidation on fission product release is being incorporated in a new transient fission product release code, FREEDOM. Release experiments in support of this code are being performed in hot cells at CRNL; eventually code predictions will be compared to the integrated fuel experiments in the Blowdown Test Facility (BTF).

Experiments to investigate the integrated thermal/mechanical response of a CANDU reactor fuel bundle when subjected to extremely degraded cooling conditions, were completed last year. These tests indicated that even at temperatures as high as 2000°C the bundle retains its basic configuration. Relocation of molten material blocked off some subchannels but good coolant access was available through the larger subchannels. This relocation resulted in less Zr surface area available for Zr/O₂ reaction and hence temperatures were lower than expected. Further experiments are being performed to confirm the magnitude and importance of this relocation effect.
Measurements of the emissivity of Zr have been made up to 1700°C. As expected, formation of a black oxide layer increased emissivity significantly. However, the eventual formation of white oxide is expected to reduce emissivity so experiments are continuing.

4. FUEL CHANNEL THERMAL AND MECHANICAL BEHAVIOUR

Experiments are being performed to study the development of temperature gradients which are formed around the pressure tube, (i) as coolant level drops due to boil-off, (ii) due to fuel elements contacting the pressure tube or (iii) due to the pressure tube straining and contacting the calandria tube.

These tests are performed on a representative fuel channel assembly that is about 2 metres long. The heater is in the form of a 37-element bundle, with the central rod acting as the heater support. The other 36 elements have a central heater which is then surrounded by an alumina insulator located inside a pressure tube/calandria tube section which is in turn placed in a water tank to simulate the effect of the moderator.

The results of these tests are being used to improve and verify various models that have been developed to predict the formation of temperature gradients under various accident conditions. This is important as it relates to the demonstration of pressure tube and hence channel integrity under various postulated conditions.

Experiments to investigate the contact conductance between the pressure tube and calandria tube for non-conforming or wavy surfaces showed that small waves formed during the deformation of the pressure tube were responsible for the discrepancy with the results from integrated (full scale) tests. An analytical model has been devised and validated.

The effect of localized contact between the bearing pads that are brazed onto the fuel element, and the pressure tube is currently being studied, as is the heat transfer from the molten zircaloy flowing from the fuel to the pressure tube under LOC and LOECI conditions.

The CHAN III version of the CHAN II code to describe fuel channel performance during severe fuel damage accident, is being finalized and compared against experiments. This code has a more detailed fuel element and bundle model than that contained in CHAN-II.

Experiments to measure the pressure tube circumferential temperature gradients, as well as fuel temperatures and hydrogen production, are being performed and used to validate safety analysis codes. The conditions necessary to induce pressure tube rupture have been defined, and confidence in the applicability of the relevant codes has been gained.
A series of experiments were conducted over the last two years to investigate the response of a calandria tube following a pressure tube rupture under conditions typical of CANDU reactor heat transport system conditions. These tests studied the effect of internal pressure, coolant subcooling, calandria tube thickness and pressure tube crack propagation velocity. The experiments indicated that the pressure pulse increased with increasing subcooling, and a small amount of plastic deformation on the calandria tube substantially decreases the pressure pulse.

Tests are also underway this year to investigate calandria tube response following a pressure tube rupture under high temperature accident conditions. These tests will study the possibility of calandria tube dryout by condensing steam following rupture at various pressures.

5. FISSION PRODUCT CHEMISTRY AND TRANSPORT

The effect of gamma-radiolysis on the volatility of aqueous iodine species under reactor accident conditions has been examined using iodide, iodine, iodate and periodate solutions containing methane.

The computer code MAKSIM, to calculate the behaviour of iodine under reactor accident conditions, is being adapted to run on an IBM/XT computer.

The solubility and vapour pressure of many linear, branched and aromatic organic iodides have been measured as a function of temperature. This data is used to calculate the aqueous/gaseous partition coefficient of iodine.

Design of the radiiodine test facility (RTF) is completed and its construction initiated. The construction is expected to be completed by summer 1987. The chemical behaviour of iodine in water and gas with gamma radiation fields will be studied using the RTF. Tests are scheduled to start in the summer of 1987.

Tests are being performed on various chemical filters to study their effectiveness of iodine absorption and retention as a function of adverse conditions including the effect of poisons. The effect of absorbed sulphur dioxide, methyl, ethyl, ketone, and ammonia on the adsorption of methyl iodide by TEDA-impregnated charcoal has been investigated. The effect of nitrogen dioxide on TEDA-impregnated charcoal and ammonia or KI-impregnated charcoal were also investigated.

The chemical behaviour of semi-volatile species in the primary heat transport system is being investigated. Mass spectra of the vapours in equilibrium with Cs$_2$Te and Cs$_2$TeO$_3$, at 1200°C, have been measured using the mass spectrometer–knudsen cell apparatus. The measurements were used to derive vapour pressures and Gibbs energies of vapourization. The chemical speciation of ruthenium was studied using equilibrium thermodynamic calculations.
Construction of a facility to study liquid aerosols is completed and commissioning tests are under way. Experiments to study creation, transport and removal processes will start in summer 1987.

6. CONTAINMENT BEHAVIOUR

The area of concentration in this program is the study of hydrogen combustion behaviour. The program is intended to investigate two basic areas; these are containment integrity and pressure response, and the integrity of components in containment.

Using the double-kernel method, the laminar burning velocity database for hydrogen-air mixtures has been extended to lean mixtures down to 8% hydrogen. In the turbulent burning velocity experiments, a database of burning velocity as a function of fan-induced turbulence and composition, has been completed.

Fabrication and assembly of the bench-scale test facility for hot surface ignition studies have been completed and the facility has been commissioned. The hot-surface ignition test will be carried out in 1987-1988. A large scale test facility is being designed to obtain a database for hydrogen effervescence from water to predict the rate of outgassing from the moderator during postulated accidents.

Combustion experiments have been performed using stoichiometric hydrogen-oxygen mixtures to study the effect of obstacles leading to a transition to detonation and the limit conditions have been correlated with the detonation cell size of the mixture. A high-speed, spark-schlieren photographic system was developed to record the transition to detonation in these tests. Further test on the transition distance are being carried out to arrive at criteria for transition to detonation.

Detonation cell sizes have been determined in a 15 cm diameter tube for stoichiometric hydrogen-oxygen mixture with added diluents such as steam, helium, carbon dioxide and helium-steam mixtures. In the absence of diluents, the detonation limit was the same as the flammability limit. For mixtures with diluents, the cell size increases and approaches the tube diameter well before the flammability limit composition. An experimental apparatus with a larger-diameter tube is under consideration to ascertain whether the detonation limit is the same as the flammability limit for these mixtures.
LWR Safety Research in Czechoslovakia: An Update

by Miroslav Hrehor
1. Status of the Czechoslovak nuclear energy programme

At present Czechoslovakia is operating 7 nuclear power reactors on two sites: 4 units at Jaslovske Bohunice and 3 ones at Dukovany totalling 3080 MWe. The 4th unit at Dukovany is in commissioning and will be put into operation by the end of this year. Another 4 units of the same tape and output are under construction at Mochovce. Recently, construction work has started at a new site - Temelin, where four 1000 MWe units are planned. Three additional new sites for nuclear power plants with two WWER 1000 units at each one are under evaluation and preparation.

Table 1. Nuclear Power Plants in CSSR

<table>
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<tr>
<th>Site</th>
<th>NPP</th>
<th>Power/MWe/</th>
<th>Stage</th>
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<tr>
<td>Jaslovské Bohunice V-1</td>
<td>2x440</td>
<td>in operation</td>
<td></td>
</tr>
<tr>
<td>Jaslovské Bohunice V-2</td>
<td>2x440</td>
<td>in operation</td>
<td></td>
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<tr>
<td>Dukovany</td>
<td>EDU</td>
<td>4x440</td>
<td>three units in operation 4th unit under commissioning</td>
</tr>
<tr>
<td>Mochovce</td>
<td>EMO</td>
<td>4x440</td>
<td>under construction</td>
</tr>
<tr>
<td>Temelin</td>
<td>ETE</td>
<td>4x1000</td>
<td>work on site</td>
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Within the period to 2000, the new plants will not only cover increased demand for power, but they will also substitute for supplies from some coal-fired steam plants that are supposed to be closed down.

The share of elektric energy produced in nuclear power plants will thus exceed 50% of total output of energy in the country, representing about 18% of all primary energy sources.

WWER - 440 reactors representing the basic generation of Czechoslovak nuclear power plants are to be gradually put into operation until 1992. The first reactor of the new generation of WWER - 1000 should be put into operation in 1991 and programme should go on with these reactors. By the year 2000 five of them should be in operation and one nearing completion. Speeding up this programme is under consideration.

Experience acquired in operating WWER 440 particularly proves their high reliability. Due to this feature, they contributed considerably to the stability of the electricity supply system, particularly in periods of extraordinary climatic conditions.

2. Nuclear safety policy in CSSR

In spite of satisfactory results, the issues of nuclear safety are under permanent review. Research results in this field at home and abroad are used and measures are taken to improve nuclear safety.

Compliance with safety regulations, which have been issued in accordance with the IAEA Safety Series is controlled by the State Nuclear Safety Supervision of Nuclear Installations. This function was entrusted by the act No. 28/1984 on the State Nuclear Safety Supervision of Nuclear Installations to the Czechoslovak Atomic Energy Commission, a central body of the state administration not directly involved in the construction and operation of NPPs. Subsequently to the act No. 28/1984 the Inspector General and nuclear safety inspectors were appointed.
On sites Jaslovské Bohunice and Dukovany special Nuclear Inspectorates with inspector-residents were established.

Czechoslovak licensing procedure is based on deterministic approach. Our reactor safety concept is fundamentally in agreement with international standards and as such it is based on defence in depth with three consecutive levels. As a design basic accident a double ended primary coolant pipe break at any location with unrestricted escape of coolant is used for the purpose of safety analyses.

Three level safety reports containing the analysis and the quality assurance programme of the power plants (ordering safety report, preliminary safety report, preoperational safety report) are released successively in accordance with the intention to build, to construct and to operate nuclear power plant.

Nuclear safety requirements concerning activities taking place on the Czechoslovak territory are fixed in generally binding legal regulations.

Table 2. Nuclear Safety Legislative Basis

<table>
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<td>Act No. 28/1984</td>
<td>on the State Nuclear Safety Supervision of Nuclear Installations</td>
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<tr>
<td>Regulation CSAEC No. 2/78</td>
<td>on Ensuring Nuclear Safety During Design, Licensing and Construction of NPP</td>
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<tr>
<td>Regulation CSAEC No. 4/79</td>
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Regulation CSAEC No. 9/85 on Ensuring Nuclear Safety of Research Installations

Instruction CSAEC No 1/85 on General Principles of Personnel Training, the Procedure of Issuing Certificates Authorizing to Handle Equipment and the Examination Statutes.

Regulation CSAEC /draft/ on Ensuring Nuclear Safety During Waste Management

3. LWR Nuclear Safety Research

The intensive nuclear programme in Czechoslovakia is accompanied by adequate R&D activities. The institutional structure as well as the composition of research and development projects correlate with the needs and tasks of organizations playing decisive role in the manufacturing, construction and operation of the NPPs.

Table 3. Main Safety Related State R&D Projects

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<td>Nuclear Power Plants with WWER 440 and 1000 Reactor Types</td>
<td>Federal Ministry of Metallurgy and Heavy Industry</td>
<td>Škoda Concern, Plzen</td>
<td>To meet the needs of the manufacturers of NPP components /mastering the know-how/</td>
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<td>II/</td>
<td>Start-up and operation of NPP with WWER 1000 units</td>
<td>Federal Ministry of Fuel and Energy</td>
<td>Nuclear Power Plant Research Institute, Jaslovské Bohunice</td>
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PURPOSE: To respond plant specific start-up and operational issues and provide input to the utilities in their preparation for safe operation

III/
TITLE: Nuclear Safety of NPP with LWR
SPONSOR: Czechoslovak Atomic Energy Commission
CONTRACTOR: Nuclear Research Institute, Rez near Prague
PURPOSE: To provide the technical basis for rulemaking and regulatory decisions, to support licensing and inspection activities

IV/
TITLE: Waste Management
SPONSOR: Czechoslovak Atomic Energy Commission
CONTRACTOR: Nuclear Research Institute, Rez near Prague
PURPOSE: To ensure long-term reliable and safe waste disposal

V/
TITLE: Utilization and Reliability Improvements of Technological Equipments of NPP with WWER 440
SPONSOR: Federal Ministry of Fuel and Energy
CONTRACTOR: Nuclear Power Plant Research Institute, Jaslovske Bohunice
PURPOSE: To improve operation and maintenance management of NPPs and to respond basic questions of plant aging and backfitting

VI/
TITLE: Diagnostics and Reliability and Life-time Assessment of Nuclear Installations
SPONSOR: Czechoslovak Atomic Energy Commission
CONTRACTOR: Nuclear Research Institute, Rez near Prague
PURPOSE: To realize modern diagnostic methods and to increase operation reliability and life-time of selected equipments and components.
4. Current state of selected topics of LWR Safety Research

4.1. Transient Analysis

The analysis of the facility response on the postulated malfunction must provide assurance that the safety features engineered into the plant are adequate to mitigate the consequences. A lot of different computer codes were developed or implemented to solve such problems as:

- pump coastdown
- turbine trip
- feedwater transients
- valve performance and phenomena
- natural circulation etc.

Among them the Soviet system codes DYNAMIKA and MOST 7 are widely used for modelling and predictions of plant system behaviour in steady state and transient conditions. Paralelly our codes SIDY/MOD 1, D-1 and STAMOD 1000 P are used. Codes VVER-D...
REPA 1D enables to simulate the processes with time-dependent reactivity disturbances. The pump coastdown can be analysed using new version DYMO VVER code.

Using this code, many practical problems have been solved as a part of pre- and post-experimental analyses of new units. Development of reliable analytical tools that provide a realistic prediction capability for the NPP systems under a variety of normal and off-normal conditions continues. Number of developments are oriented toward improved numerics.

4.2. Loss of coolant accident

The results of the analyses of the dynamic and thermal-hydraulic effects during large break LOCA accident form an important basis for evaluation of the advanced concept for improved safety of WWER reactor type.

Two computer codes SHOCK and BAREL are used for calculation of dynamic action on reactor internals during blowdown. The thermal-hydraulic analysis is carried out by Czechoslovak codes LENKA and SICHTA 85/MOD1, Soviet code TEČ-M, US code RELAP 4/MOD5 and 6 and German code BRUCH-D. The aim of thermal analysis of the containment is to provide a picture of the pressure-temperature behaviour of the air-water-steam mixture. A fundamental set of computer codes for this analysis is formed by the complex TRACO V.

While the development of analytical tools for blowdown phase is nearly complete, results of heat transfer calculations during refill and reflood phase are still undergoing refinement. Codes NORCOOL-1 and RELAP 4/MOD 6 are used for these types of calculations.

Following the TMI accident, much of the focus for reactor research was redirected toward small break scenarios and small break response including the complex phenomena associated with two-phase circulation. An improved computer code SLAP-2 was used for pre and post-experimental analysis of small LOCA experiments prepared at KFKI Budapest on PMK-NVH facility.
4.3. Fuel behaviour in accident conditions

In 1985 an integral model of the fuel element represented by the code PIN was improved. The test calculations demonstrated that the code PIN can predict with sufficient accuracy the temperature distribution in fuel and fission products release from the fuel. A modular computer code FRAS released in 1984 is intensively used for thermohydraulic modelling of fuel behaviour during LOCA accident. The code consists of following modules:

FRATEMP - non-stationary temperature distribution in fuel element including heat generation and oxidation of cladding material
FRAMECH - structural analysis of cladding including fuel-clad contact
FRAGAS - axial pressure distribution in the gas gap
FRAPLEN - non-stationary temperature in the gas plenum including heat generation due to radiation
FRAHYD - hydrodynamics in fuel channel including two-phase flow
FRALFA - heat transfer coefficient from cladding into coolant
FRAGAP - heat transfer through cladding-fuel gap
FRSTAT - mechanic state equation of cladding material

Values of all material properties are automatically read from MATRO library.

For special strain-stress analyses of cladding material during LOCA accident two additional codes RUPTUR 85 and DEFOR 85 are used. For two dimensional stress/strain analysis of fuel element including creep and plasticity a foreign code AXIDEF is under implementation. Implementation and testing of SSYST 2 code continues.
4. Dynamic response of structures

For the purpose of an evaluation of the WWER-1000 containment resistance against the extreme external loads a new version of CEFRA-IMPACT code is used. For modelling of steel-concrete structure during impact-loads a special models REBAR and CAPI are under further development and testing.

4.5. Probabilistic safety assessment

Foreign and domestic computer codes have been modified and developed as a tool for practical use of probabilistic methods. Starting with fault tree methodology the first experience with probabilistic approach was gained in many practical exercises.

Among modified and developed codes the following are the most widely used:

**SAFEDO 2** - direct simulation code using Monte Carlo techniques able to calculate unavailabilities taking into account the periodical testing of two groups of elements

**ALLCUT - 2** - modified US code ALLCUTS for qualitative and quantitative analysis of fault trees

**KADO** - the Bulgarian code of analytical type which evaluates the unavailability of a system on the basis of the Vesely's "Kinetic tree theory"

**SPOL - 3** - Czechoslovak code based on an analytical method. It enables effectively combine the manual and computer elaboration of fault trees
LOTR - 3 - original CS code for calculation of the instantaneous and average unavailability of a system by an analytical method taking into account periodical testing

CAT PREP KITT - computer code complex for automatic construction of fault trees.

FANTIC III - US code for quantitative fault and event tree analysis

RANGE, COSMOS - foreign codes for uncertainty analysis

In recent years, the reliability evaluation programme of reactor protection and safety-related systems promoted by the State Nuclear Safety Supervision of the CSAEC has been initiated.

Under this programme, very detailed plant specific fault trees were developed and reviewed in close co-operation with designers, operation and maintenance personnel. Detail maps of system functions such as
- emergency core-cooling system (PS, HPIS, LPIS)
- spray and barbotage systems
- emergency feedwater systems
were developed and following reliability analyses have been performed:
- emergency power supply of primary circulation pumps
- reactor control and protection system
- fire protection in cable spreading rooms
- electrical supply for instrumentation and control system.

The current content of the safety reports does not include and does not require in general the use of probabilistic risk methodology (PRA). Despite of this the event tree method has already been used with success in some safety reports. For example, the failures of main circulation pumps, the rupture of the main charging and main steam collectors, interruption of delivery of charging water, etc., have been analysed.
All these studies provided the better understanding of interdependences of sub-systems and components and identified the dominant components and events that were the main contributors to the unavailability and unreliability of the systems.

As a result of these analyses some components were duplicated (motor-operated valves, level indicators), some operator information had to be monitored (position of valves, airlock blocking), instruction for the operator were supplemented, periodical tests were also supplemented and made more stringent (e.g. in the case of the flowmeters), some measures against the possibility of the common-cause failures were accepted etc. These analyses revealed possibilities to increase or to assure the reliability of the safety systems with no additional great effort and cost.

The analyses carried out did produce a very important feedback to the plant operation. It was thus confirmed that this type of retrospective analysis is a must if one wants to update and improve safety of the plants.

Experience shows that nuclear safety can be ensured only by closely linking advanced technical safety systems with high level operation of the technological equipment. The attention is paid to close monitoring of operational experience as the best assurance of keeping operational safety high and the only way of verifying system reliability analysis results.

In spite of the fact that reliability analyses have become in Czechoslovakia practically routine, probabilistic approach as a tool to assist in safety assessment of nuclear power plants is still a developing art. Problems exist in many areas, for example, non-adequate data base, complex of phenomenological questions associated with severe core damage, human factor, common cause failures, uncertainties etc.
REACTOR SAFETY RESEARCH IN FINLAND IN 1987

L. Mattila

STATUS REPORT FOR THE IAEA TECHNICAL COMMITTEE ON THERMAL REACTOR SAFETY RESEARCH, VIENNA 9. - 12.6.1987
REACTOR SAFETY RESEARCH IN FINLAND IN 1987

1 NUCLEAR ENERGY STATUS

Finland has currently 4 nuclear units in operation producing about 35% of the electricity supply in the country. The demand for power has increased steadily and, hence, decisions on building new large power plants are needed.

In fact, a comprehensive national electric power construction program for the next ten years or so, including 500-1000 MWe of new nuclear capacity, was at an advanced stage of preparation in the Finnish government as the Chernobyl accident took place. Political and popular reaction to this accident has resulted in a new situation. It seems that decisions on new nuclear capacity have to be postponed for several years. Capacity additions needed for the first half of 1990's will be based on peat and coal power plants.

There have also been scattered requests for a premature closing of the present nuclear plants. The requests have, however, not gained any strong political backing.

The process of revising the nuclear energy law has finally passed the main obstacles. In 1986 the law was finalized and passed by the Parliament. The law has to be passed also by the newly elected Parliament. It is believed that this will be done promptly during this year. The new law will effect an extensive modernization of the nuclear legislation. The law will give the Parliament the final say on building new nuclear installations in Finland.
2 OVERALL VIEW OF REACTOR SAFETY RESEARCH

The Finnish Atomic Energy Commission has appointed an expert group consisting of representatives from utilities (IVO and TVO), the Finnish Centre for Radiation and Nuclear Safety (STUK) and the largest research organization (VTT) to assess the nuclear energy R&D needs for the next five years or so. Table 1 is an interim list produced by the group for the current research priorities.

Table 2 summarizes the content and volume of publicly funded research projects in 1986 having pronounced reactor safety aspect. The table also lists participation in international co-operation. Most of the work is carried out in VTT and most of the funding comes from the Ministry of Trade and Industry and from VTT's budget. The utilities and the safety authority (STUK) also contribute to a couple of the projects and some joint Nordic funds for the so called NKA programs are also available.

3 UPDATE ON SELECTED REACTOR SAFETY RESEARCH TOPICS

Severe accidents and source term

The utilities were directed by STUK in a letter in June 1986 to submit programmes aimed at identifying and installing additional systems for severe accident mitigation - like filtered containment venting - before the end of 1988. Both utilities have submitted their programmes, outlining R&D work, technical solutions, and proposed implementation schedules.

Most of the research in the area of severe accidents and source term is carried out within the Severe Accident Assessment (VARA) Project at VTT, in progress since 1983. Research is closely integrated with the practical application needs of the utilities and STUK.
VTT, usually together with the utilities and/or STUK, participates in several international research projects such as LACE, LOFT and IDCOR and the Nordic source term research programme NKA/AKT. Furthermore, bilateral co-operative research is carried out with EPRI, the Swedish RAMA project and the Kernforschungszentrum Karlsruhe. VTT also takes part in the work of several working groups of the OECD/NEA/CSNI.

Laboratory scale experimental research is carried out at Kuopio University. The first phase of a research program on aerosol hygroscopic behaviour is finished and the second phase has been initiated. The effect of various parameters, such as relative humidity, primary aerosol size, temperature and residence time, on aerosol growth is being investigated.

The main tool for plant specific analyses is the integrated plant systems code MAAP from IDCOR and the consequence analysis code ARANO developed at VTT. Models of the MAAP are benchmarked with several more detailed mechanistic codes. Some development work has been needed to modify MAAP applicable for the Finnish plants. Recently, a study of selected accident sequences up to environmental consequences for the TVO plant (Asea-Atom BWR) with the MAAP and ARANO codes was finished. A similar study for the Loviisa plant (VVER-440 PWR) is under way.

Chernobyl related research has comprised, in addition to extensive environmental measurements, mainly
- reactor physics calculations of the RBMK reactor,
- reliability calculations of core cooling systems of the RBMK reactor and
- development of environmental consequence analyses methods as part of the Nordic NKA/AKTU research program.

Safety analysis tools

In the framework of an extensive research programme on "Numerical simulation of processes" VTT has started together
with IVO the development of an advanced process simulation system (APROS). This system will be based on the use of stable and efficient solution algorithms, fast calculation of material properties, and efficient use of data base techniques, vectorization and parallel computation. The system will also include a CAD-type operator interface to allow easy definition and construction of new models and comprehensive presentation and handling of simulation results.

Based on this system a prototype nuclear plant analyzer, so called Modular Plant Analyzer (MPA), will be built. The main advantage in this system compared to conventional analysis codes will be cost-effectiveness and easiness of use, like the interactive control of calculations. The prototype will be ready at about the end 1980'ies.

**Operator support systems**

Modern information technology, like the application of the artificial intelligence and expert system techniques, provides new and efficient means to construct operator support systems for normal operation and the handling of disturbance and accident situations in nuclear power plants. VTT participates in a Nordic co-operation program on "Advanced information technology" (NKA/INF, 1985 - 1989) addressing the use of these new technologies for the construction of operator and safety organization support systems. A couple of prototype systems will be constructed and recommendations and guidelines developed.

Software quality is a key question in using computerized systems in nuclear power plants. This question has been studied in a co-operation project with the OECD Halden Reactor Project since 1976. This work has included the development of a formal specification language (X), interactive analysis and support system (SPEX) and software evaluation and testing methods. In the future evaluation and
validation methods of expert systems will be addressed.

New thermal hydraulic test facility

The design of the new PACTEL- (Parallel Channel Test Loop) facility at Lappeenranta University of Technology (LTKK) for SBLOCA, LBLOCA and abnormal transient investigations was started in 1986. The construction of the components is under way. The facility is a 1/305-scale simulation of the Loviisa VVER-440 PWR. The design of the facility is based on power and volume scaling preserving the component elevations. The maximum pressure is 8.0 MPa.

The primary circuit of the VVER-reactor is simulated with a U-tube construction corresponding the pressure vessel of the reactor and with three primary loops, each containing a steam generator and a hot and a cold leg loop seal. The PACTEL-facility has also a pressurizer, accumulators, HPI- and LPI-pumps.

The core in a cylindrical shroud has three separated parallel fuel bundle channels each having 48 fuel rod simulators. The full-length fuel rod simulators are in the actual fuel rod geometry: heated length 2420 mm, rod outer diameter 9.1 mm and lattice pitch 12.2 mm. The maximum power of the core will be 1 MW, which represents 22 % of the nominal power of 144 fuel rods in the reference reactor.

The three identical steam generators of the facility each simulate two steam generators of the real plant. The pressure characteristics of the HPI- and LPI-pumps and the pressurizer are preserved, only the flow rates and the volumes are scaled down.

PACTEL-facility will be the next step in a series of thermal hydraulic test loops built and operated jointly by LTKK and VTT. The earlier REWET facilities have been used, for example, for comparisons between different ECC injection
alternatives (cold leg/hot leg/combined), to study the effect of spacer grids on rewetting and, currently, for natural circulation investigations.

**Leak before break**

The applicability of the Leak Before Break principle in the design of the large nuclear plant primary pipes and pressure vessels is being assessed. An overall survey of all aspects (leak detection, loads, break development, piping response, inspection, probabilistic assessment, etc.) has been completed. A large amount of small size pipe testing has been carried out as well as a destructive test with a reactor vessel size pressure vessel. Two further experiments with large vessels are planned for 1988. Extensive code development and post-test analyses are under way.

**Ageing**

The specific ageing mechanism related to materials degradation are neutron radiation embrittlement, corrosion and fatigue loading. Preventive maintenance requires thorough understanding of the processes as well as development of means to monitor the process. Increased emphasis of research and development work is being focused on these areas. A new topic is the utilization of reliability techniques in combination of materials, NDT and water chemistry knowledge for assessing the remaining life-time of nuclear power plant piping.
APRIL 1987

REACTOR SAFETY RESEARCH PRIORITIES IN FINLAND

- PSA
  - Better assessment of the safety level of the existing plants: Level 1 PSA’s are under way, extension to Level 2 (or even 3) considered
  - Safety requirements for new plants

- Severe accident management
  - Identify and mitigate phenomena threatening the containment
  - Containment integrity and failure modes
  - Source term and ways to reduce it
  - Long-term accident management

- Transient and accident analysis
  - Nuclear plant analyzer
  - Code validation, particularly for VVER-plants

- Plant ageing and degradation
  - Mechanical systems
  - Electrical systems
  - Instrumentation
  - Control room

- Human aspects
  - Operator support systems, including the use of AI
  - Emergency operating procedures

- Transient frequency reduction by improved operational procedures
  - Optimization of preventive maintenance and testing
  - Optimization of safety related technical specifications
  - Computer supported plant status monitoring and maintenance
  - Fire safety

- Emergency preparedness
  - Information and communication systems (plant, planning area, national, international)
  - Consequence prediction models

- Safety features of new plants
  - Advanced light water reactors; increased use of passive and inherent safety features
### PUBLICLY FUNDED REACTOR SAFETY RESEARCH PROJECTS IN FINLAND 1987

**May 1987**

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### ABBREVIATIONS:

- **FMI**: FINNISH METEOROLOGICAL INSTITUTE
- **HY**: HELSINKI UNIVERSITY
- **KY**: KUPIO UNIVERSITY
- **LTKK**: LAPPEENRANTA UNIVERSITY OF TECHNOLOGY
- **STIK**: FINNISH CENTRE FOR RADIATION AND NUCLEAR SAFETY
- **VTT**: TECHNICAL RESEARCH CENTRE OF FINLAND
THE NEED FOR FURTHER NUCLEAR SAFETY RESEARCH
A PRESENT FRENCH VIEW

by J. Bussac, Director of Nuclear Safety and Protection Research
presented by R. Zammite, Reactor Safety Research Co-ordinating
IPSN, France

A - GENERAL CONSIDERATIONS

If the severity of the Chernobyl accident moved public opinion to distrust nuclear plants safety — temporarily, we hope so — all those who have responsibilities in the nuclear field, are challenged to infer lessons from it. They have to use every possible means in order to avoid a severe accident in nuclear power plants and most particularly to avoid radioactive releases above the level which could threaten the health of surrounding populations.

As mentioned in the INSAG report written last September, the Chernobyl accident was caused by "a remarkable range of human errors and violations of operating rules in combination with specific reactor features which compounded and amplified the effects of the errors and led to the reactivity excursion".

This emphasizes the importance of:
- safety in design (incentive to more inherent safety),
- strict safety organization and formal operating procedures,
- human factor and man-machine interface (cf. excessive confidence of operators, lack of indicators revealing the precarious state of the reactor to them),
- outstanding role of the containment building.

.../...
Indeed, there is nothing really new since TMI. Nevertheless, we should give more consideration to:
- reactivity insertion (for example: rod ejection) and ATWS,
- steam explosions,
- human factors,
- containment performance.

Finally, the main lesson to be learned from the Chernobyl accident is that a severe accident with radioactive release is far more than simply a topic for theoretical research and risk studies. Such accidents may really occur. And nobody can claim that it is truly impossible for a very severe accident to occur in other countries, even if the measures taken do insure against such high radioactivity releases.

We must not rely on the present situation and decrease our efforts concerning matters of safety: this is vital for the future of nuclear energy.

We can improve the safety of our plants. The first way is the NPP operational feedback. This is one way, perhaps the most important today, but there is another way, research, for the following reasons:

1 - On events that are already listed (for instance thermal-hydraulic transients or mechanical fatigue), there is a need for better understanding and extensive scientific knowledge.

2 - Anticipated information is required for aging problems and for the evolution of technology (advanced concepts, development of automation...).

3 - Chiefly, for severe accident issues, we cannot rely on experience feedback which only gives us the possible initiating events. To provide adequate protection and — and this point is most important — to be ready to manage severe events, we need the physical understanding, as complete as possible, of the different scenarios given by the PRA, which are worth examining. In particular, detailed procedures suitable for accident management can only be set up by careful analysis and full understanding of the phenomenology of the possible events. This is the role of research.

When planning experiments, we have to be careful that they are not primarily aimed at demonstrating directly a result for an actual reactor issue, because this demonstration cannot be obtained as a general rule, even through sophisticated experiment: Research is aimed at improving scientific knowledge, not at demonstrating a special result for authorities.
Experiments are either of the analytical or of the separate effects types (in order to set up physical models) or they are integral tests which can be used as benchmarks to validate analytical procedures and computer codes. In the latter case, however, few experiments are generally available because of their cost whereas we need for these tests as many configurations as possible, because of the great number of parameters involved so as to be able to apply or extrapolate the models to a realistic reactor scenario, in the right way and with a good level of confidence.

To follow on from these general considerations, the French nuclear situation is characterized by:

1 - A vigorous nuclear program offering with the NPPs already in operation more than 70% of our electricity requirement.
2 - A high degree of standardization: from the 49 NPPs in operation at the end of 1986, 43 are PWRs: 33 in the 900 MWe class and 9 in the 1300 MWe class.
3 - The maturity of our PWR safety research program: when FRANCE launched its PWR series, we first used U.S. safety criteria. Our objectives were then to evaluate the safety margins included in the design. Gradually we developed our own programs so as to answer to specific questions raised by our analysis, to increase the overall competence of our Protection and Safety Institute and to develop our own installations.

B - MAIN RESEARCH TOPICS

To go into the main topics of PWR safety research in FRANCE, while devoting special attention to the recent evolution which we might discern since the Chernobyl accident, things can be arranged in three categories according to the action to be taken:

1. Those which are unchanged; or,
2. Those which have been accelerated since the accident; and,
3. New lines of thinking after Chernobyl.
1.1 In the first category, the field of thermal-hydraulics led on to the CATHARE code, a two fluids best estimate code. The first version is operational since October 1985, following vast and in-depth qualification performed with a lot of analytical experiments. We are now beginning the operation of BETHSY, a 3 MW integral loop, built in cooperation between CEA, EDF and FRAMATOME, aimed at:

- Analyzing several types of transients and degraded core accident sequences;
- Completing the validation of the computer codes: CATHARE as well as foreign codes in the case of cooperative agreements;
- Checking the operator procedures, especially those referred to as "H procedures" which are at the limit of the design basis conditions and deal with corrective actions needed to prevent core uncovery or to reflood the core.

1.2 The second main field of investigation is that of fuel damage, in particular in the PHEBUS test reactor. After completion of the LOCA type tests, we are now beginning the severe damage phase III (up to 1,850 MW) and phase IV (up to 2,500 MW) tests which are planned to last for at least three years more. These tests have been designed in order to improve the understanding of physical phenomena governing PWR core degradation under different prototypical conditions, and to provide a set of "benchmarks" to validate SFP computer codes. For these goals, several sets of parameters (temperature, pressure, power, steam and hydrogen flow rate) have been selected to form a "grid" and to allow a systematic approach for evaluation.

1.3 Other fields are those concerning:

- Phenomena important to containment performance such as hydrogen stratification and the effects of defragmentation or detonation;
- Characteristics of the aerosols produced by means of out-of-pile tests HEVA carried out on irradiated fuel heated at 1,800 °C in an induction furnace;
- Study of aerosols in the primary system and in the containment, deposition and remobilization of these aerosols, especially in damp atmospheres;
- Behavior of iodine compounds in a radiation field, CsI radiolysis and reactions with the paintwork on the walls, or with concrete.

.../...
1.4 The laboratory tests carried out in Cadarache, now completed, have led to the specification of the sand filters which it has been decided to install as a vented system on all the EDF PWRs. This cheap (2 MF) and simple filter (gain of a ratio of ten over released aerosols, noble gases aside) is part of the ultimate procedure "U 5" intended to limit any dangerous pressure rise inside the containment building in a scenario such as that of the delta mode in the WASH 1400 terminology.

2 - To come now to the research work which has been accelerated since the Chernobyl accident, three different research areas deserve mention:

2.1 The first relates to the venting system: this device is now considered very urgent (and all the PWRs will progressively be equipped with them as of mid 87, at a rate of 2 per month with 1 filter for 2 units), and so important in fact that we shall very probably run next year at Cadarache a test on a full size spare filter unit in order to get an accurate performance evaluation and provide a verification of the system.

2.2 Before Chernobyl, we were preparing a future PHEBUS program known as the "Phebus Fission Products" which implied upgrading of the facility to run a series of integral experiments in the field of source term assessment. This will require the use of pre-irradiated and re-irradiated fuels. Our objective is to provide a representation of all the phenomena, from melting of a fuel assembly to release from a simulated containment, including transport out of the primary circuit and the stratification, deposition and remobilization effects within the containment, using actual fission products in a variety of PWR accident scenarios. The aim is to verify the adequacy of the present models with the data of these experiments, the non-omission of essential phenomena and to provide prototypical benchmarks for the validation of advanced systems of codes for source term assessment.

While discussions for cooperation are presently underway with the Commission of the European Communities and other possible foreign partners, the Commissariat à l'Energie Atomique has decided to include in its 1987 budget commitments to order the fuel and several investments in order not to delay this program considered as very important.
2.3 The third program, already launched and which has been pushed ahead, is RESSAC, a study of measures to be taken in the event of off-site contamination, especially regarding the recovery of contaminated soils for return to normal conditions of living.

3 - Finally, some new lines of research considered in FRANCE in the last few months have been submitted for discussion in preliminary form to different French committees for which a post-Chernobyl review has been required:

3.1 Ultimate coolability of a largely degraded core is still under question today. Some data can be expected from planned PHEBUS Phase III and IV experiments but this will probably not be enough.

3.2 An experiment is under consideration in Grenoble to look into the coolability of melted Corium penetrating from the reactor pit into the base mat: this would be an out-of-pile experiment with artificial heating: can we put water on the Corium? Could there be a dangerous steam explosion?

3.3 Also under consideration is the interaction of concrete with actual Corium, that is, with irradiated fuel elements, more particularly to analyse the final interaction phase, when the Corium is freezing. This is difficult to obtain from experiments being performed with simulated materials and external heating. At such a time, what is the rate of progression of Corium and of gas production?

3.4 In addition, with the aforementioned RESSAC program, we decided to study the diffusion in the soils and the transfer to plants and animals of the main radioactive species present in some selected French locations, due to Chernobyl fallout: some contaminated zones, although far from being dangerous, are comparable to extraordinary laboratories for observation and measurements.

3.5 What we primarily intend to do now is to make a vigorous effort, by collaboration between EDF and CEA, to extend our set of robots for inside and outside intervention. We need radio guided or teleguided robots for remote handling as well as for civil works.
3.6 Several other measures are being considered for the case of crisis management: in the radiological field, sophisticated means of radiological site survey, such as gamma scanning of contaminated zones from a helicopter, improved measurements of radioactivity in the plume and a centralized data collection system are needed. In the sanitary field, a method for rapid sorting of irradiated people, treatment of men injured by radiological burns, medical population survey etc., are what is called for.

CONCLUSION

1. The Chernobyl accident confirms rather than invalidates some of the main safety research priorities settled after TMI on:

a) Primary importance of the human factor with:
   * Operator training
   * Man-machine interaction
   * Safe operational organization
   * Sound procedures
b) Accident management
c) Containment performance.

2. Besides immediate actions for improved communication with the public and with the media, and a fresh review of the previous assessments of our nuclear plants in the light of issues underlined by the Chernobyl accident, the main conclusions drawn in FRANCE after Chernobyl have led to the following actions:

a) Accelerated safety research which is still needed, especially into severe accidents and source term issues and increased international cooperation in that field.
b) Preparation of the means capable of dealing with an emergency situation and efficient crisis management, with significant amount of related R and D.
REACTOR SAFETY RESEARCH

M. V. Banaschik

June 9-12, 1967
Within the last years it has become clear that the worldwide effort to develop a technology required to generate electricity using nuclear energy has been successful. However, the projected extensive use of nuclear power has led to widespread public concern over the hazards to health posed by those radioactive materials associated with nuclear power generation. Unlike fossil-fuelled power generation, which uses fuels known to man for a long time, nuclear fission is a recently discovered phenomenon.

In the Federal Republic of Germany 11 pressurized water reactors (PWR) and 7 boiling water reactors (BWR) are in operation. They contribute about 36% to the overall production of electric power. At present, 3 PWRs are under construction, one prototype gas-cooled high-temperature reactor (THTR 300) are in the commissioning phase. A small gas cooled high temperature reactor (AVR) has been in operation since 1967. The construction of the prototype fast breeder reactor SNR 300 is almost completed. A small fast breeder reactor KNK II has been in operation since 1973.

The utilization of nuclear energy in a country as densely populated as the Federal Republic of Germany requires high safety standards. The principal aim of reactor safety is to protect the man and environment against a release of radioactive materials contained in nuclear reactors. This protection is accomplished by using separate passive barriers to contain radioactive materials and engineering safety features to ensure operational safety.

The passive safety barriers are:

- fuel cladding
- pressure vessel and piping
- reactor containment.

The design basis of the safety concept for nuclear power plants comprises a well-balanced combination of engineering safety features:
- a proven design, comprehensive quality assurance and control measures during component manufacturing and plant construction
- in-service inspection during operation
- engineered safety equipment for limitation of consequences of incidents aimed to control both the spectrum of potential incidents as well as to prevent fission product release.

The main objective of this safety concept is to prevent any accidents in nuclear power plants which may lead to releases of radioactive fission products into the environment.


Since then, the Federal Republic of Germany, has carried on reactor safety research within the scope of a Program that has continuously been updated and has unlike in most other nuclear energy countries, always been independent of current licensing and supervisory problems.

The general objectives of the safety research Program in respect of increasing utilisation of nuclear energy are the following:

- a continuous extension of the knowledge of potential causes and sequences of accidents;
- a continuous further development of the methods used for a realistic assessment of safety;
- the analysis and evaluation of safety margins;
- the further development and optimization of safety technology.
The research and development works of the reactor safety research program concentrate on the four basic research areas:

- Component Safety and Quality Assurance
- Plant Transients and Accident Sequences
- Man/Machine Interaction
- Risk and Reliability

Component Safety and Quality Assurance

The objective of this area is the prevention of accidents and safe carrying of operational and accidental loads.

The research results gained to date improved the understanding of quality degrading influences, i.e. trace elements, crack formation and toughness degradation.

Independent from the component safety quality assurance may be considered as a redundancy to prevent component failures. It consists of sufficient preservice and inservice inspection as well as of adequate detection systems to monitor leakage and vibration of components and loose parts within the primary circuit. The high performance of these surveillance systems has been demonstrated in several plants.

As regards inservice inspections, research investigations were performed to develop nondestructive testing methods which are able to detect flaws at a very early stage.
These investigations included the development of remotely operated manipulating systems in order to improve positioning and to decrease the radiation burden on the inspecting personnel.

National as well as international Round Robin tests i.e. Defect Detection Trial (DDT); Program for the inspection of Steel Components (PISC) demonstrate that the defect detection capability of nondestructive testing systems used today for the inservice inspection of components of the latest German LWR's is nearly 100%, even for small defects (corresponding to 3 mm diameter of the equivalent reflector size), which are not important to safety.

Future research work aims at improving the understanding of long-term defect mechanisms including the operational loadings such as temperature, pressure, corrosion and irradiation. These aspects become more and more important with increasing plant lifetime.

Moreover questions concerning aging of components and plant-life extension have to be answered with respect to safety.

In the field of nondestructive testing research will be continued to intensify the possibilities for recognizing and to size defects. This is necessary in order to understand the relationship between real crack extension and the crack-opening mechanism, i.e. crack-initiation, -propagation and -arrest and finally to find out the crack opening-time-function under real loading conditions.

Plant Transients and Accident Sequences

On the basis of a large experimental program the efficiency of the German ECC-System could be demonstrated for all possible break configurations within the primary circuit. Taking into account the fuel rod behaviour the maximum fuel rod temperature has been much lower than postulated by the licensing procedure. The in-
vestigations concerning the large break accident will be fina-
lized by the 2D/3D experiments performed in cooperation between
the BMFT, USNRC and JAERI. Emphasis is put on the investigation
of multi-dimensional phenomena in the upper plenum and the ef-
fects on emergency core cooling.

Recent tests results confirm the importance of multidimensional
effects and offer deeper insight into the processes. The emergen-
cy core cooling concept of German PWR's, especially the effectiv-
ness of the hot leg coolant injection has been confirmed.

New computer codes were developed, which realistically predict
power reactor transients. The analytical efforts were accompa-
nied by experimental investigations. For small leak initiated
incidents the influence of the steam generator secondary side
is of major importance for the removal of residual heat from the
reactor system. It has been proven that for German pressurized
water reactor design sufficient heat removal can be maintained
in these situations. This is valid even if natural circulation
is interrupted and the presence of non-condensable gas restricts
heat transfer from the primary to the secondary side. In LOBI
and PKL test facilities the designed cool down of the secondary
side by a gradient of 100 K/h was demonstrated to be feasible.

Further research work regarding transients will concentrate on
a still better understanding of initiating events, courses, conse-
quences and of possibilities for operator interactions. The
results will serve for code improvements and the elaboration of
intervention procedures.

With respect to the subject of severe accidents (core meltdown
accidents), comprehensive investigations are being carried out
focusing on PWR scenarios by means of analytical and experimen-
tal work. The results furnish the information basis for analyti-
tical risk investigations as well as for analyses of potential possibilities of intervention in the case of severe accidents.

It was demonstrated by large-scale experiments within the DEMONA Project that the concentration of fission product aerosols in the containment atmosphere is reduced by a factor of approx. 10,000 within a short period of time (7-30 h). The BETA experiments on the corium/concrete-interaction showed that this interaction does not lead to any (essential) further release of active fission product aerosols from the molten core and that the initially rapid penetration of the molten core into the concrete involves a considerable temperature decrease of the molten mass.

The research area Accident Management gains more emphasis. Using this term we think of activities to systematically utilizing additionally implicit safety reserves of the plants as demanded in the Federal Republic of Germany and proven in licensing procedures for those events which had not been considered explicitly by design. These existing safety reserves are only partly self-active; partly they are to be initiated. During beyond-design-basis event occurrences, safety measures can be initiated and carried out to end or mitigate the effects of event sequences by applying the operational and safety systems.

In order to explore the existing safety potential of a plant as given by the conceptual design of the plant, it is necessary to have as realistic a picture of the plant behaviour under abnormal conditions as possible. This knowledge forms the basis for diagnosing plant conditions and derivating full symptom sets as well as for deducing measures to recover or replace disturbed safety functions. In order to create those assumptions, realistic analyses on plant behaviour for a spectrum of events are to be performed if not already existing.
The extent required, the efficiency of the specified measures and the existing or necessary physical boundary conditions have to be examined analytically and, the load reduction by the existing systems has to be evaluated. Proposed measures to prevent core damage should be simulated in the calculations by models. An additional aim of the analyses is to determine at what latest time these measures are to be undertaken or within which maximum time span they are to be carried out.

For the purpose of Accident Management it is of paramount importance to know the safety margins of the relevant plant on a realistic basis and quantitatively, to have available best estimate codes, and to know as realistically as possible the accident courses and their temporal sequences.

The most important problems with respect to severe core damage are formation and distribution of hydrogen, its burning behaviour and the resulting loads on the containment. Questions with respect to fission product retention in the containment and to the source term are other subjects.

Planning efforts concerning accident management may become even more important. Investigations and studies have to be focused on measures involving the use of operational equipment or additional systems simple to install in order to cope with the consequences of possible severe accidents.

Better planning for accident may exhibit a considerable potential for the further reduction of the residual risk and for mitigation of the consequences of accidents.

**Man/Machine Interaction**

The operation of nuclear power plants and operators responses to perturbations are important to safety. This situation is underlined by probabilistic analyses as well as international operating experience.
This experience reinforces again the approach taken in FRG to assist the personnel in complicated situations by automatic safety systems (the 30 minutes concept) and to provide the operator with extensive training and with sufficient information on the status of the plant, including the automatic systems. Limitation systems protect the plant by means of soft actions before the safety systems are activated, so that the overall plant protection becomes even more failure tolerant.

Consequently in recent years much of our research work has been concentrating on the man/machine interaction for situations preceding a serious disturbance. These developments became possible by taking advantage of the tremendous advances in microelectronics outside the nuclear field.

Additional sources of information utilizing stochastic signals are exploited in the early failure detection systems, e.g. vibration monitoring.

Yet another way of providing more information on the plant status is the online application of simulation models. E.g. the core surveillance system derives detailed information on the power distribution and its likely development using fast simulation models. Such information is of particular interest in boiling water reactors where local power density is being monitored.

In the meantime a new line is being explored, where the latest developments in the field of informatics - artificial intelligence methods - are being considered to make the knowledge of designers and experts, accumulated over the years, available to the operator.

Common to these developments is the use of modern display techniques and programmable microelectronics. A precondition for the introduction of such systems into the control rooms, is the proof that they are reliable enough to fulfill the stringent requirements of nuclear grade components. Therefore, another line of research deals with the reliability of software and the required qualification.
In addition to this more technically minded approach to further improve the man/machine interaction, research is also directed towards the human element itself and the operator training.

Since in the control rooms hardly any disturbances can be witnessed, the research is extended towards simulator studies. Teams of operators at full scope simulators are tested. The main problems of this kind of research are the huge amount of data collected even for short simulator sessions and the question of sensitivity because of the low probability of human errors.

I'd like to point out that this whole research area has been pursued in several countries so that active international cooperation has been possible even though approaches concerning the distribution between manual operation and automatic controls may be different.

The OECD Halden project has been a focal point of research in the field of man/machine interaction and we expect that the good international cooperation established so far will be continued.

Risk and Reliability

Phase A of the German Risk Study, shows that accident-related risk of nuclear power plants is relatively low. The Study suggested important plant improvements and showed where major emphasis was needed in research. Moreover, it is also the basis for possible contributions of an advanced safety technology and other measures to a further mitigation of risks.

Presently Phase B of the Risk Study is under work. The investigations are being executed in more detail. Refined methods and physical methods are used with take into account the present status and progress of safety research as well as an improved data base and advanced operating experience. Up to now the results of phase B
show that the consequences of severe accidents had been overestimated in phase A. Furthermore the results show that the consequences can further be reduced if measures are taken to prevent containment failure due to overpressure and hydrogen combustion.

In order to improve probabilistic analyses, realistic models and methods have to be at hand. Therefore the analytical methods and models will still be improved to reduce the uncertainties. Especially models for considering human and common cause failure have to be developed on and dynamic reliabilities to be introduced.

Summary

Summarizing my statement I would like to state, that more than 14 years of nuclear safety research in our country have resulted in a considerable amount of important results.

They have improved and confirmed the safety concept of German pressurized and boiling water reactors in principle.

Moreover they enabled the development of powerful tools for safety evaluations and continuous improvements of the safety technology.

Nuclear safety research will continue to accompany the peaceful use of nuclear energy in Germany. This will take place on the basis of

= operational experiences (worldwide)
= a continuous deepening of knowledge about courses and consequences of accidents
= probabilistic safety analyses.
The main medium range future activities will be

- extension of knowledge on material behaviour under long-time operation

- man machine interaction

- improvement of understanding about the contribution of human factors to the nuclear risk

On the long range we have to deal with:

- further improvements of safety technology and of tools for safety assessments

- keeping available know how and technical capacity to react on new questions

- safety research for advanced reactor concepts.

Nuclear safety is an international issue. So, as in the past, we consider it highly important, to execute all research tasks in cooperation with the nuclear community. Thereby, we, together with our partners, take further on advantage of

- enlarging the field of information

- evaluating results on a broad basis

- exchanging experiences

- discussing new research activities

and thus care for additional assurances of safety aspects and contribute to the achievement of an international consensus in safety issues.
AN UPDATE ON THE SAFETY RESEARCH ACTIVITY
IN HUNGARY

by
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To be presented at the Technical Committee Meeting
on Thermal Reactor Safety, Vienna, 9-12, June 1987.

Budapest, Hungary

1987
INTRODUCTION


In 1986, a new five year plan was started in Hungary. For its time period, 1986-1990, a coordinated national research Programme was initiated containing four different subprograms, as follows:

1. Optimal operational conditions of the units being in operation;
2. Nuclear safety and environment protection; /see in Annex/;
3. Fundamental R+D works;
4. Development of equipments for NPP's.

The R+D programme has been sponsored by several ministries and the industry.

The main features of Hungarian nuclear programme are the following:

--- there are three units in operation and one in start-up phase /VVER-440 type each/;
--- nuclear programme will be continued with VVER-1000 type reactors;
--- share of the nuclear energy production from the total one was 23% in 1986.

The national coordinated research programme has been reviewed in the light of Chernobil-accident and has been slightly modified. In the future, even more emphasis will be given to the probabilistic safety analysis and severe accident analysis, while analytical and experimental R+D works will be continued for "until design basis accidents" as well.

The main objective of the programme is to assure the safe operation of the Paks NPP, protection of the environment against the eventual adverse effects of the nuclear energy. The immediate results from the completion of the project should be
a development in the safety analysis of the NPP. Further advantages as the support of the activities of regulatory authorities, improvement of emergency planning can be anticipated.

This report includes short summaries in research fields as:

- Plant transients and SBLOCA;
- Severe accident analysis;
- Probabilistic safety assessment activity;
- Man-machine interaction;

1. PLANT TRANSIENTS AND SBLOCA

Two experiments and computer code analysis are presented. The experiments were performed on the PMK-NVH integral type test facility. The experimental programme, the main features and the capability of the test facility are presented at the 4th and 5th TC-SR Meeting in 1985 and 1986. The computer code analysis were performed by the RELAP4/mod6 at the IAEA in the framework of the "Regional Programme for Computer-Aided Safety Analysis". The plant transient analysis presented in this report is a total loss of flow, the SBLOCA is a 7.4% break in the cold leg. The latter was the subject of the 1st IAEA Standard Problem Exercise.

1.1 Plant transient analysis of the PMK-NVH: total loss of flow

The experiment modelled is a total loss of flow incident in the Paks Nuclear Power Plant with the initial conditions and sequence of events given for the plant. The question is whether the safety of the plant is satisfactory in case of this type of incidents. The experiment was performed on the PMK-NVH test facility, which a scaled model of the Paks NPP. The experiment was followed by a RELAP4/mod6 computer code analysis. A detailed code analysis based on the experimental data was performed by the COBRA-3C code to assess the minimum values of the critical heat flux ratio,
The main conclusions are as follows:

-- The RELAP4/mod6 is a suitable tool for the analysis of loss of flow incidents in VVER-type NPPs.
-- The minimum value of the critical heat-flux ratio is higher than 2.9.
-- In case of station black-out incident the safety of a VVER-440 plant is satisfactory.

1.2 Analysis of a 7.4% cold leg break

The first standard Problem Exercise /SPE-1/ organized and conducted by the IAEA was successfully finished. The final meeting of the SPE-1 was held in Vienna December 1986. Seventeen organizations from 12 countries participated and performed 20 test predictions. The comparison report is under publication at the IAEA.

The test was performed on the PMK-NVH facility with the main features as: cold leg break modelling a 7.4% break in the NPP starting from full power as nominal operating conditions; without injection from hydroaccumulators; with HPIS corresponding to the case when only one of the three pumps is available; secondary side is isolated after transient initiation; for the sequence of events plant data are used.

Conclusions accepted by the final meeting are as follows:

-- The PMK-NVH facility is a suitable tool for code validation experiments and for the investigation of basic phenomena occurring in the accidents of NPPs. The instrumentation and data acquisition system can be considered adequate for code assessment purposes.
-- The experiment selected for the SPE-1 was appropriate, but very demanding and sensitive from the modelling point of view, causing a spread in fuel temperatures.
-- No general conclusion on optimum code or model is reached, because code assessment is a long process involving several steps.
SPE-1 was one of the steps and considered to be highly successful, therefore IAEA and Hungary intend to proceed their joint efforts for conducting the 2nd SPE.

1.3 The 2nd IAEA -PMK-NVH Standard Problem Exercise /SPE-2/

On the basis of the good experiences of SPE-1, the 2nd Standard Problem Exercise is going on. The experimental basis is the PMK-NVH facility. The experiment chosen by the final meeting of the SPE-1 is a test modelling a 7.4% cold leg break again, but with hydroaccumulator injection.

On the basis of the first trial-like experiments, the SPE-2 is much more rich in events offering good possibilities of code assessment for Member States interested in.

2. SEVERE ACCIDENT ANALYSIS

Analysis for beyond DBA's have been carried out with the assistance of the IAEA that made it possible to become familiar with the source term code package and to run the codes on the IAEA-computer. These first results have been very preliminary because a lot of input data arbitrarily presumed were used. Unfortunately there is a break now in running the codes because of temporary lack of financial resources, but a suitable solution is being looked for continuing the programme. Some works have been carried out in the hydrogen field. Hydrogen generation during a hypothetical severe accident has been assessed and its removal possibilities have been discussed. Further investigations are needed for the final decision.

3. PSA - ACTIVITY

Works on system reliability analysis /SRA/ and probabilistic safety assessment /PSA/ in nuclear field was started in the late seventies. Initial activity was concentrated on development and adaption of tools and methods for SRA and PSA purposes.
In 1985 Hungary joined to the IAEA’s Inter-Regional Project on PSA /INT-9/063/ and initiated a national coordinated R+D project to perform level-I analysis of NPP Paks. The overall objectives of this project are to setup a plant specific failure and reliability data bank, as well as to produce a probabilistic model of VVER-440 type nuclear units. Based on plant data and model the following studies are planned:

-- calculation of core melt /CM/ frequency, ranking of systems and components through determination of their importance to CM frequency;
-- investigation of operational procedures, maintenance and test policies to improve operational safety.

Detailed milestones for the PSA programme were setup within IAEA coordination and formulated during an expert mission /20-24, January 1986/.

To determine numerical CM frequency, six dominant initiating events were related. Up to the present small and large LOCA event sequences were analysed and partial CM frequencies were calculated.

To improve operational safety, the optimal testing frequency of high pressure ECCS was determined and the revision of maintenance policy for condenser cooling system /raw water system/ was carried out.

4. MAN-MACHINE INTERACTION

4.1 Full-scale simulator

In order to speed up the development of the full-scale simulator, Hungary has bought Finnish know-how and the simulator is constructed under Finnish supervision. This simulator will be an exact replica of the control room of Paks NPP, Unit 3. The Technical Design of the simulator was finished by the end of 1985 and at present the off-line model development is ready.
The Control Room of the Simulator is also ready and it was successfully connected to the Control Room Interface System this winter. The development of the plant computer is finished and its commissioning tests were concluded this spring. In this way the whole environment is ready at present for the on-line model tuning and in-house testing. Some basic parameters of the full-scale simulator are as follows:

- number of measurements: 3500
- number of valves: 1400
- number of pumps: 150
- number of switches: 500
- number of controllers: 100
- number of protections: 1200
- number of different malfunctions: 50

The extent of process and instrumentation system simulation is sufficient to allow the simulator to be operated according to the plant operating instruction for every normal plant situation and for the simulated disturbance cases. All major operating procedures needing control room actions may be carried out with realistic responses on the indicators and display units.

4.2 Basic principle simulator

The WWER-440 basic principle simulator will be used for preliminary operator training at low cost, so that an instructor can select the most promising operator candidates for further training in the full-scope simulator. In this simulator the control room is replaced by a control desk with indicators of the most important process variables and with a control station. The operation of the plant and its dynamic behaviour are simulated in the power range during steady-state and transient situations. The research and development of this simulator is a part of the Coordinated Research Program of the IAEA, and as the Hungarian contribution to this program,
all of the models of the WWER-440 basic principles simulator will be submitted to the IAEA in the form of national reports.

At present the following models are ready:
--main technological units of the primary circuit;
--main technological units of the secondary circuit;
--main controllers of the primary circuit.

The controller models of the secondary circuit is under development.

The control desk was designed in a modular way which allows changes to be made in a flexible manner. On the desk there are three main areas, as:
-- mimic diagram with alarm windows;
-- analogue indicators;
-- control station.

This desk is a microcomputer-controlled special input/output peripheral of the simulator computer. At present the output part of the desk is ready and the input part is under development.

Some basic parameters of the basic principles simulators are as follows:
-- number of analogue measurements: 96
-- number of digital parameters: 400
-- number of valves: 35
-- number of pumps: 12
-- number of controllers: 28
-- number of protections: 20

The basic principle simulator provides the dynamics of the main power-plant processes and basic controllers in the 5-110% power range. Because of the scope of the instrumentation and control simulation is considerably smaller than that of the full-scope simulator, this system is much less expensive than the replica simulator.
4.3 SPDS

Some progress in the works for safety parameter display system /SPDS/ has been achieved. Sensors and cables and other components of the measuring channel located inside the containment and being foreseen to be used for SPDS have been investigated under simulated accidental environmental conditions. Failure modes of the components and their capability for further operation and modification of their characteristics have been investigated experimentally. Further demonstration of the two-level SPDS with simulated accident sequences will be possible on IBM PC and the most convenience display formats will be decided by involving operational personnel.
Annex

The main topics of the national Coordinated Research Programme in the nuclear safety field in Hungary /1986 -- 1990/

1. Computer code developments for simulation of accidents until DBA's and analysis for different LOCA's and transients. Experimental verifications.

2. Evaluation of the main equipments from the point of view of safety. Stress analysis and fracture mechanical investigations for different mechanical and thermal load situations under normal operation and DBA's.

3. Containment analysis in cases of DBA's. Thermohydraulic and stress calculation, radioactivity releases to the environment.


5. Emergency planning evaluation programme /real-time running codes, environmental measurement systems, simplified methods, etc./

6. PRA—programme
   -- severe accident analysis /core-melt process, degraded core, steam explosion, corium behaviour, containment response, source term/;
   -- level - 1 PSA;
   -- optimal testing and maintenance policy;
   -- investigation of operational procedures;

7. Man-machine interaction
   -- training simulator;
   -- SPDS.

8. Environmental protection /dose-response correlations, radioactive waste disposal, personal dosimetry, etc./.
IAEA Technical Committee on Thermal Reactor Safety Research

VIENNA - June 9-12 1987

ITALIAN NUCLEAR SAFETY RESEARCH PROGRAM IN 1986
ITALIAN NUCLEAR SAFETY RESEARCH IN 1986

Present paper is aimed at shortly updating information on Italian nuclear regulatory research programs presented at the last IAEA Technical Committee on thermal reactor safety research held last year in Vienna. It will mostly report on R&D activities directly promoted or utilized by Italian nuclear regulatory authority (ENEA/DISP), while more extensive research activities, dealing with nuclear safety and health protection, are performed by other ENEA Units such as the Environment and Health Protection Department, the Thermal Reactor Department and the Basic Intersectorial Technologies Department. Above quoted ENEA Units play the main role in meeting ENEA/DISP safety research needs.

ENEA expenditure commitments, specifically devoted in 1986 to nuclear safety and health protection research programs, can be evaluated to be about 45 thousand millions lire, while additional safety related R&D activities are performed by ENEA in the framework of prototypes development and industrial promotion.

The whole 1986 ENEA R&D activity has been focused on the lines of accident prevention, consequences evaluation and mitigation, probabilistic methodologies development and utilization. The majority of performed activities was a continuation of programs previously started and already mentioned in 1986 report.
1. Structural integrity

Research activity aimed at assuring the integrity of the reactor pressure boundary has been carried on along the two main lines, already followed during former years, dealing respectively with pressure vessel and piping integrity.

As concerns pressure vessel integrity particular efforts were devoted to an experimental program at Pisa University for thermal shock tests.

As part of the program on pressure vessel integrity, analytical and experimental activities on crack propagation under dynamic loads, were pursued in collaboration with CISE, as well as analytical and experimental studies on impact phenomena were still carried on in collaboration with ISMES.

Significant results were achieved during 1985 and 1986 in the assessment of neutron embrittlement of pressure vessel materials. The analytical correlation, developed in 1985, aimed at the prediction of the irradiation damage of structural steels, has been improved, within ENEA/DISP, basing on larger data base.

An experimental program on neutron damage has been started using compact specimens irradiated in test reactors.

In 1985 and 1986 the research program on piping integrity was mainly oriented toward stainless steel pipes, covering in particular, the fracture behaviour of weldments and HAZ.

A research program on residual stress evaluation is in progress in collaboration with CISE. The program includes both analytical and experimental activity and is aimed at the development of two elastoplastic finite elements computer codes (ETERNO and AMENO).
The codes will assess temperatures and residual stresses distribution by simulating fabrication welding processes and heat treatments.

An experimental program has been performed at Garigliano N.S., now in decommissioning, aimed to optimization of measures methods of containment leakage.

2. **LOCA and transients**

A relevant part of Italian nuclear regulatory research effort has been devoted, during 1986, to LOCA and transient analysis; both analytical and experimental activities were performed. The commissioning of the two integral test facilities, already mentioned in last year report, is presently concluded and the natural circulation tests are now in progress.

The PIPER-1 International Standard Problem exercise ISP21, proposed and accepted last year in the frame of CSNI, will be performed in the last quarter of 1987.

As concerns the SPES facility, which simulates (scale 1:427) the PWR Italian standard nuclear power plant an International Standard Problem exercise, note as ISP 22, has been proposed in the frame of CSNI. The test should be a double-blind loss of feed water with E.F.W. delayed and should be performed in the first quarter of 1988.
During 1986, as during former years, ENEA has been involved, directly and through collaboration agreements with other national organizations, in the pre-test and post-test analysis of LOBI experimental program.

An experimental program has been performed on PWR Steam Generator thermo-hydraulic behaviour during accidental transients. Two test facilities was utilized: CFA-FREGENE (water-freon loop) for the over-all steam generator performance during variations of secondary inventory; ARAMIS (air-water loop) to study the steam separator behaviour under accident conditions (steam line break).

In the frame of the collaboration agreement with USNRC on thermal-hydraulic research, ENEA is participating to the International Code Assessment and Application Program (ICAP) which purpose is to obtain a considered view of the accuracy and validity of USNRC thermal-hydraulic codes and their range of affordability.

3. **Plant-environment interaction**

   The great effort devoted by ENEA to seismic studies and research during former years has continued during 1986 substantially along same headlines.

   The following activities related with plant-environment interaction are, in particular, to be mentioned:
   - studies in the field of hystorical and instrumental seismicity;
   - development and management of seismic arrays;
- integrated research on seismo-tectonics aiming at providing an even deeper knowledge of the characteristics of the national territory with particular reference to specific areas selected or suitable for nuclear installations;
- research on the safety of LWR containment systems, including in particular, both analytical and experimental activity on hydrogen production-combustion-distribution and control in accident conditions;
- testing of filtering systems of gaseus radioactive effluents under heavy environmental conditions (IPF 8000 test rig reported in 1984 report).

A research program is presently under way, in collaboration with Rome University, to study the mechanical behaviour of the soil under static and dynamic loads; the specific objective of the program is to investigate on possible reliable and simple correlation between static and dynamic characteristics which could be of use in the process of soil geotechnical characterisation.

4. **Severe accident (S.A.) analysis and risk evaluation**

A significant effort is devoted by ENEA and particularly by ENEA/DISP to study and research activities in this area. During 1986, analysis has been performed as application to the Italian nuclear plants of the Severe Accident criteria developed during 1985 (in the containment post core melt accident scenarios, if credit is given to operator actions also in presence of defects of the containment isolation system, releases of the most volatile
fission products (noble gases excluded) may be contained within the 0.1% of the core inventory with a probability greater than 95%.

A more complete study for Italian NPPs source-term definition for regulatory purposes is presently under performance.

Besides to the participation in the USNRC multiannual research plan on S.A. (the agreement with NRC on Severe Fuel Damage research program has been reviewed for 4 years starting from 1987), we participate in the EPRI-LACE program performing pre-test and post-test calculations and related support studies. The program is aimed at experimentally investigating on aerosol behaviour inside the containment buildings of LWR under postulated S.A. situations, and at providing a significant data base for the validation of aerosols transport computer codes.

In order to make more realistic accident evaluations, SPARTA Project (Suppression Pool Aerosols Retention Testing Apparatus), aimed at investigating fission product pool scrubbing effectiveness, is in progress.

This experimental work is being developed in C.R.E. Casaccia Laboratories. The construction of the test facility will be completed in the first quarter of 1988.

Program objectives will be:
- scrubbing decontamination factors measurement
- model validation and development
- code assessment.
The major milestones will consist in the evaluation of the decotamination factors acted by a large scale pool through a full scale X-quencher and horizontal vent devices. Several tests will be carried out in a small facility in order to facilitate the study of pool retention in the saturated condition.

5. **Chernobyl accident evaluation activities**

Some days after Chernobyl accident, ENEA has produced a dossier which reported all the actual available information on the accident. This document has been adjorned about each two-weeks in order to give the possibility to critically analyze all the material in real time.

The final version, at the end of May 1986, contained the following topics:

- Plant description and possible accident sequences.
- Information about radioactive release measurements and diffusion trajectories.
- Description of Italian safety network.
- Generals about possible radiological consequences.
- Many appendices describing Italian nuclear reactors, design features, differences with the Russian type. Other chapters were devoted to radioprotection general problems.

After Vienna Post-Accident Review meeting, some analytical researches has been set up, and are now in progress, in order to try to have a better understanding of the accident sequence.

The activities were related to the use of some thermohydraulic and neutronic Codes to predict Chernobyl power behaviour.
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The Laguna Verde Nuclear Power project (LVNP) has prompted the development of research activities aimed at obtaining further information on specified safety aspects of the project. Among these aspects are those related to the performance of the plant under normal and off-normal conditions and the verification of the basis for the emergency planning.

The LVNP consists of two 650MWe units with BWR reactors and Mark II pressure suppression containments. The first unit is 97% complete and is expected to load fuel by June, 1987. Pre-operational testing is now proceeding. The second unit is now 45% complete, civil works have reached the refuelling pools level and it is programmed to start commercial operation by 1990.

The scarcity of resources both human and economic, inherent to the state of development of the country and to the prevailing restrictions derived from our huge foreign debt, limits the reactor safety research activities in such a way that, apart from some original development in some isolated and very specialized topics, these activities consist mainly of the implementation and slight modifications if any, of analysis techniques developed in the industrialized countries.
Accordingly, the projects which are described below have been undertaken with the purpose of fulfilling an existing need derived either from the necessity to obtain a better knowledge of the plant characteristics (to be employed for example, in the assessment of the adequacy of the emergency operation procedures) or from specific concerns expressed by the regulatory body or by other involved groups.

LOCA ANALYSIS

During the past year the computer code RELAP 4/MOD 5 has been implemented by the regulatory body and efforts are presently directed to its applications in the analysis of a double-ended pipe rupture. So far, a good representation of the steady state has been obtained and some preliminary runs of the transient case have been made. The results show that some parameters have to be better defined in order to remove the inconsistencies detected.

The TRAC BF1 code has very recently been obtained from the USNRC Office of Regulatory Research and its implementation in the available installations is planned to be completed by the end of this year. It is felt that this code, developed specifically for BWR plants, will provide a closer representation of our plant. Work with this code will be undertaken by the regulatory body and the institute for electric research.
As reported last year, the station black-out for the Laguna Verde Nuclear Power Plant is a safety aspect of special interest for the Mexican Regulatory Body (CNSNS) in view of the need for further information on systems performance under the existing grid stability conditions. The CNSNS undertook a probabilistic safety assessment of the Laguna Verde Nuclear Power Plant under black-out conditions which has been developed independently of the level 1 PSA of the plant, currently in progress.

At present, the fault-tree analysis for five front-line and support systems as well as for the AC-DC power distribution system has been concluded. The event-tree analysis is at an early stage and it is expected to be concluded in the first quarter of 1988.

Preliminary analyses were performed with the codes FTAP (cualitative analysis) and IMPORTANCE (cuantitative analysis) implemented in a microcomputer, with the MIFTA system described below. The SETS package is operative in a mainframe computer and is being used for the analysis of sequences.

USE OF MICROCOMPUTERS IN PSA

A fault-tree analysis system for a PC XT/AT microcomputer (MIFTA) has been developed by adapting publicly available
mainframe computer codes (FTAP, IMPORTANCE) and developing some auxiliary software. The fault-tree analysis system is conformed by a database system, a qualitative and quantitative fault-tree analysis package, and a graphic editor.

FTAP was modified to produce the fault-tree logic input to SETS code for the sequence event analysis. All options of the original FTAP code are supported and have been tested. Sensitivity to the data and determination of dependent events can be automatically performed for use in fault-tree analyses and event sequences.

LEVEL 1 PSA OF THE LAGUNA VERDE NUCLEAR POWER STATION

As previously reported, a probabilistic safety assessment of the Laguna Verde Nuclear Power Plant was initiated in the fall of 1986. This project is a joint effort of a group of national institutions, which includes the state-owned utility, two research institutes, and the regulatory body.

The analysis is being performed using and input processor for the version of the FTAP code implemented in a VAX 11 minicomputer.

At present, the fault tree analysis for six front-line and support systems has already been conclude. According to the revised plans, during the present year the whole work will be
completed and the final report will be issued in the first quarter of 1988.

ASSESSMENT OF THE SOURCE TERM FOR THE LAGUNA VERDE NUCLEAR POWER PLANT

This project has been recently undertaken by the regulatory body to obtain a specific source term for the Laguna Verde Nuclear Power Plant (LVNPP) with state-of-the-art techniques and codes.

Once the specific source terms are available for various LVNPP severe accident sequences they will be used to assess the performance of the safety systems, the time-evolution of the accident and to quantify the release to the atmosphere and the consequences. The results will also be used to assess the adequacy of the emergency plan as regards to the size of the emergency planning zone.

The schedule of the project reserves the rest of this year for the procurement of the source term package, its implementation in a CDC computer and acquaintanceship with the codes that form the package.

FUEL MANAGEMENT

The implementation in a UNIVAC 1100 computer of the SCANDPOWER package for fuel management purposes (FMS) has been accomplished
by our utility's nuclear energy department with the collaboration of the institute for nuclear research. The package has been validated using the core parameters of the Hatch-1 plant, and the incorporation of the Laguna Verde values into the FMS is now proceeding. This work was sponsored by the IAEA and as a continuation, a project consisting in the acquisition and implementation of the RAMONA code for transient analysis has been initiated by the department.

Acknowledgements

This compilation was prepared from the information provided by the personnel involved in the listed projects. Thanks are given to E. del Valle, J. A. Becerra, M. Schwarzblat, J. Valdés and C. Villanueva, leaders of the several projects and to the researches that work under their directions.
THERMAL REACTOR SAFETY RESEARCH
IN THE NETHERLANDS

Contribution to IAEA Technical Committee Meeting
Vienna 9-12 June 1987

K.J. Brinkmann
The current situation in The Netherlands

There are two power stations in operation:
Borssele, a 460 MWe PWR since 1973
Dodewaard, a 65 MWe BWR since 1969

Up to the year 2000 decisions are needed on ± 6000 MWe generating capacity and the questions are what is possible and what is desirable. The first question deals with safety and radioactive waste deposition for scenarios including nuclear capacity. The second question addresses the strategic mix between nuclear, coal, oil and gas. Following a public debate in 1982, the Dutch government came by the end of 1985 with a standpoint according to which at least 2000 MWe nuclear capacity should be added with the provision that the feasibility of a final safe deposition of nuclear waste should be shown. A common construction bureau was founded by several utilities. The specifications for the new power plants should be ready in May 1986. Site selection procedures were started for approval by parliament also in May 1986. However the Tsjernobyl accident on April 26 initiated amongst others a discussion in parliament on the implications for the Netherlands. It was decided to postpone further activities and to evaluate the consequences and the impact of the Tsjernobyl accident and to reflect again on the future use of nuclear energy. The government issued a plan addressing several issues to reflect on. One issue was "the safety of nuclear power plants". Here the main topics for further activities were addressed in a governmental note published in October 1986:

1. Development and establishment of safety rules and guidelines;
2. Safety studies of the existing nuclear power stations;
3. OSART missions to those stations;
4. Update of information on the source term to enable decision making on siting of new power stations.

1. Safety rules

Safety rules for applications in The Netherlands had been developed in The Netherlands by amending the IAEA Codes and Guides on Design, Quality Assurance and Operation. This effort was near completion when the Tsjernobyl accident occurred. Prior to the formal establishment of these criteria and guidelines they will be scrutinized for possible further changes or amendments that can be derived from the Tsjernobyl accident. The results can be expected in the course of 1987.

2. Safety studies of existing plants

The objectives of the safety studies of the nuclear power plants in Borssele and Dodewaard are to ensure that occurrence of phenomena that have led to the Tsjernobyl accident can be prevented and to investigate if the containment function in the event of a severe accident can be assured.
Reactivity transient will be reviewed for various operational states together with reactivity control provisions, interlocking andadministrative procedures.

In the area of fire prevention, detection and mitigation the plants will be subjected to a review in order to establish if they meet the state of the art measures.

A "human factors" review of the plants will encompass a study of possibilities to prevent automatic safety function performance inclusive fast shutdown, an assessment of (emergency) operating procedures, inventory of instrumentation and controls available to the operator to follow (severe) accidents and the environmental qualifications of these instruments, an assessment of procedures for maintenance, testing and experiments in the plant, a review of control room habitability and an investigation of the information on the operational status of safety system.

For the containment studies of the two plants, all loads will be indentified that during a coremelt accident could challenge the containment integrity. Where necessary, the studies will lead to proposals for reasonable hardware or software modification which would ensure that:

- containment integrity can be maintained, or
- the eventual loss of containment integrity can be delayed significantly, while the gross containment structure can be kept intact, or
- the probability of containment failure can be reduced significantly, or
- the radioactive releases to the environment, in the event of containment failure, can be reduced significantly.

These studies should be finished by mid 1987.

3. OSART missions

The IAEA has been invited to send OSART missions. The mission to the Borssele nuclear power plant was completed in 1986, the mission to the Dodewaard nuclear power plant was completed in May, 1987.

4. Source term study

The source term issue has always been a major topic in the discussion on expansion of the nuclear programme and the selection of sites. The Tsjernobyl accident has focussed the attention to the potential contamination of large areas over long periods. Hence, in preparation for a new discussion on future nuclear plans (which is expected early 1988) an update of the source term know-how is required. A study is being made of all relevant phenomena determining the source term. Also extra engineered safety features which have a further reduction potential will be assessed. This study will be finished by mid 1987, it is being carried out by ECN. A related ongoing study also involving ECN, addresses the possible social economic effects of a nuclear accident.

The whole exercise was planned in a way that parliament could finish their discussions before the summer of 1988. However a further postponement is expected.
II Research

Nuclear research, including fusion and fast breeder research amounts to about 40% of the total research effort at ECN. In the present absence of a further Dutch commitment to nuclear energy, the nuclear research goal is to have available an operational reservoir of knowledge with respect to relevant parts of the fuel cycle. Narrow ties with international activities and programs involve participation in OECD-Halden, LACE, USNRC-SARP, Batelle HBEP, ICAP.

With respect to thermal reactors the main research areas are:
- risk and safety analyses,
- disposal of radioactive waste,
- radiation hygiene,
- complementary projects (safe guards, basic simulator training, reactor noise analyses).

In the area of risk and safety analyses, the current main effort is, as explained in I, on the source term update and on the connected consequence analyses. Further work concerns process and system behaviour during reactor transients and locas. The main computer codes in use are RELAP V mod 2, SCDAP, STCP, CONTAIN. Experimental work concerns the determination of thermochemical data of fission products and the study for a separate effect experiment on wall condensation and related transport of aerosols to improve the understanding of local processes in a containment after a severe accident.

In the area of land based geological deposition of radioactive waste, work is done in the framework of a national program to show that there is a safe and acceptable solution for a final deposition. The work involves safety studies for the various design options, hydrological and geological research and irradiation damage in salt. Much of the work is done with the German GSF under sponsorship of the CEC. Much attention is being devoted to the analysis of thermomechanic behaviour of radioactive waste as deposited in salt mines.
INFORMATION ON RESEARCH ACTIVITIES
ON THERMAL REACTOR SAFETY IN POLAND
IN 1986 - 87

Prepared by W.Dąbek, E.T.Józefowicz

for

TECHNICAL COMMITTEE MEETING
ON THERMAL REACTOR SAFETY RESEARCH
Vienna, 9 - 12 June 1987

WARSZAW
May 1987
Introduction

The activities in thermal reactor safety in Poland are closely connected with the nuclear power units, which are built or are to be built in the country in the next years, that is WER-440 (V-213 type) and WER-1000 units, and to some extent with high flux research reactors.

The first two units of WER-440 are under construction close to Larnowiec, two others of the same type are planned on the same site. For the second NPP with 4 WER-1000 units two sites are in preparation parallely (the siting procedure in our country is performed in two steps).

The activities can be divided into following areas:
- regulations and guides,
- probabilistic safety analysis,
- thermohydraulic calculations,
- experimental activities on modelling of selected phenomena,
- transport of fission products, environmental consequences and related problems,
- accountancy of nuclear materials.

Regulations and guides

- the "Atomic Law" was established by the Parliament on April 10th 1986,
- in February 1987 the Regulation by Council of Ministers was signed on the detailed competence of National Atomic Energy Agency in the frames of which the Nuclear Safety Inspectorate is situated,
- in the near future the Regulation by Council of Ministers
on the organization, detailed tasks, and acting mode of Nuclear Safety Inspectorate is expected to be signed, as well as several execution acts of the President of National Atomic Energy Agency, concerning the licensing of personnel of nuclear power plants and other nuclear installations, the dose limits and derived emergency levels, safeguards of nuclear materials, and others.

In preparation of these regulations the IAEA documents and other international documents are extensively used.

Probabilistic Safety Analysis

The IAEA methodology (level 1) is under continuous development for NPP Żarnowiec:
- the primary version of event tree was elaborated as the starting point for systems analysis,
- the general guides for construction of fault trees were prepared (with description of labeling system, classification of components, available data about success criteria etc.),
- the programme for qualitative analysis of fault trees FTAP was implemented,
- the input data for fault tree construction are collected for several systems,
- the fault tree for accident mitigation system is under development, which is thought as a model for other trees,
- the preparation of data base has started, based first on generic data, but programmed for permanent actualization with operational data.
Thermohydraulic calculations

These calculations concerned three main problems:
- some fragments of supplier independent safety analysis of Żarnowiec NPP,
- pre-test calculations for the out-of-pile test facility WIW-300,
- calculations connected with the participation in the first IAEA PKK-NVK Standard Problem Exercise.

These calculations were reported in detail on the Technical Committee/Workshop on Computer Aided Safety Analysis in Warsaw, Poland, 25–29 May 1987 and the reader is sent to this document (Appendix).

The continuation of these activities within the IAEA supported programme is highly desirable, as the calculational abilities for safety analysis of NPP's in Poland are very limited. Much effort should be devoted to perform independent analysis of all safety systems of NPP, especially containment system with barbotage shelves.

Experimental activities

The out-of-pile high pressure water loop WIW-300 reconstruction was continued during 1986. The activity was concentrated on the LCS system, the passive lines are ready, the active are under construction. In 1987 a simulated LOCA experiment is planned with this installation containing 7 simulated fuel elements, and some other auxiliary experiments connected with the construction of the in-pile loop. The measurements of the critical heat flux in the conditions
of temporary lack of power supply to main circulation pump will be performed.

The construction of high pressure in-pile loop for safety-oriented investigations with WER type fuel is continued.

The concept of experimental stand for investigation of aerosol behaviour in the WPR-1000 type containment in accident conditions is considered.

The experimental verification of strength calculations of barbotage tower constructions of WER-440 are under way.

The different methods for environmental monitoring around the site of Żarnowiec NPP are being prepared. The atmospheric dispersion will be studied in detail, first using steady level balloons (this year) and careful mathematical modelling, next a diffusion experiment (using 200 m high meteorological mast of the plant station) is being prepared. This is because the orography and the meteorological conditions in Żarnowiec site are far from typical ones.

Transport of FP's, environmental consequences & related problems

- a simple computer code UDOT for assessment of fission products transport inside WWER-440 containment and their release to the environment during ICA conditions has been developed and tested,

- a new version of the FCDA code has been developed, which can be used for assessment of fission product release from fuel elements to the primary coolant system and transport within that system during normal operation, transient and accident conditions including cladding failures (without fuel melting),
within the framework of the IAEA Regional Programme on Computer Aided Safety Analysis (R&R/9/002) several calculations of radiological consequences of accident conditions were performed using CRAC-2 code in the IAEA Computer Center, mainly for different sequences of RBLOCA's with and without containment failure in Zarnowiec NPP. These calculations had conservative character. Continuation of these calculations is foreseen in 1998 (see also Appendix).

- the preliminary studies of the zirconium - steam reaction in accident conditions started, based on literature data.

Accountancy of nuclear materials

The methods for nondestructive measurements of nuclear materials in research reactor fuels are elaborated and used. The computerization of total nuclear material accountancy is under way. These activities are undertaken having in mind safeguard and nuclear safety as well.
Abstract

The participation of the Institute of Atomic Energy and Central Laboratory for Radiological Protection in the IAEA Programme on Computer Aided Safety Analysis is presented and discussed under the light of the Polish Nuclear Programme and needs.

1. INTRODUCTION

Participation of the Polish staff in the Agency's Regional Programme for Computer Aided Safety Analysis started in 1985, three years after the Agency had made available to other participants sophisticated computer codes installed in the IAEA Computer Center. Therefore, the polish activity in the framework of the Regional Programme is relatively small.
Three main subjects may be distinguished within the project:

- A supplier independent safety analysis of the WWER-400 nuclear power plant at Zarnowiec - first in Poland - now the first two units under construction and expected to start operation in 1991;
- Pre-test calculations for the out-of-pile test facility WIN-300;
- Calculations connected with the participation in the First IAEA PKK-KVH Standard Problem Exercise.

The Polish specialists engaged in the Programme come from the Institute of Atomic Energy /IEA/ and Central Laboratory for Radiological Protection /CLRP/.

During the last two years the basic techniques in performing calculations of large and small loss of coolant accidents with the RELAP4/MOD6 code were accumulated at IEA. The staff from the CLRP concentrates on environmental consequences due to radioactivity releases for accident conditions in WWER type of reactor using the CRAC 2 code.
2. **DISCUSSION OF THE ACTIVITY**

2.1. **Large and small break LOCA analyses**

RELAP4/MOD6 was used to perform the following accident analyses for the Zarnowiec NPP with WER-440 type reactor:
- double-ended /2 x 200%/ break located at the cold and hot leg,
- study of the TMI-type of SB LOCA,
- small-size small break LOCA due to instrument line break.

The LB LOCA loop calculations for the design basis accident conditions, as the first independent effort in the framework of IAEA Programme, was reported at a workshop held in Portoroz. Based on these results separate hot channel analyses were performed for various peaking factors and normalized power. These hot channel calculations will be used as a boundary conditions /pressure, temperature, heat transfer coefficients/ for planned fuel rod behaviour analyses with the computer code SSYS?T-2. Reflood phase of process was modelled separately with the use of REFLEX-M and RAP codes available in IAE.

As concern TMI-type of SB LOCA, the 4.5% leak from the relieve valve on the top of pressurizer was considered. A number of sensitivity tests were performed in order to:
- secondary cooling,
- degree of heating,
- timing of the behaviour of the primary and secondary systems,
- axial power burn-up.

Most of the transient analyses to the LPIC were run.

Among the features of the SIM module of the transient analysis to the LPIC brought in the Safety Injection system, to extend the main circuit capacity to extend
in order to estimate the significance of:
- secondary side conditions /constant parameters,
  cooling down with constant rate and isolation/,
- degree of availability of the HPIS,
- timing of the main circulation pump trip,
- politropic expansion coefficient for safety injection
tanks partially filled with nitrogen /adiabatic or
isotermic expansion/,
- axial power distribution /typical for fresh and
  burned up fuel rod/.

Most of calculations were continued well after
2500 sec of real time unless the critical cladding
temperature and extended core uncovering conditions
were reached earlier.

Among others the results confirmed characteristic
feature of WWER - type plants that following SB LOCA,
the SIT's may run empty for a relatively long period
of transient until the system pressure can be reduce
to the LPIS set-point pressure. The calculations
brought the problem of the need of verification of
Safety Injection Actuation Signal for HPIS and deter-
mination of suitable time moment for switching off the
main circulation pumps. It should be stressed, however,
that for more realistic estimation it is highly desired
to extend the model of pressurizer.
The first series of blowdown pre-test calculations with RELAP/MOD6 for the electrically heated water loop WIN-300 show that it is possible to describe the phenomena during transient using available input data, but the lack of experimental results related to the temperature distribution and the flow rates during the transients do not allow for a more precise comparison between the calculations and experiment. The first series of the blowdown experiments were concentrated on testing of different type pressure transducers. The next calculations should give an important contribution in the planning experimental procedure and choosing appropriate instrumentation.

The calculations performed within the IAEA PMK-NVII Standard Problem Exercise provided important experience in modelling small brake LOCA’s in WWER-type reactor. It has been realized that analyst’s skill is very important factor influencing the results obtained with RELAP code. Certain assumptions related to models parameters and system segmentation may be extremely important. Due to unexpected limitation of the Project budget the pre-test calculations could not be completed and the post-test evaluation were performed with some delay and therefore were not included into the final SPE report.
2.2. Environmental consequences due to radioactivity release

During two visits to the IAEA Computer Section in 1986 the following calculations of radiological consequences for accident conditions were performed with CRAC2 code:

1. For the NPP with WWER-440 type of reactor at the site Żarnowiec:
   - Design Basis Accident /DBA/; Large Break Cold Leg LOCA; 100% of cladding failures, 10% of core melt.
   - Large Break Cold Leg LOCA; 100% of cladding failures, 10% of core melt; Failure of containment.
   - Large Break Cold Leg LOCA; 100% of core melt.

   The calculations were performed for various meteorological conditions, evacuation strategies, and some other data.

2. For the NPP with WWER-1000 type of reactor /the second Polish NPP/:
   - Large Break Cold Leg LOCA; 100% of core melt; Failure of containment.

   The calculations were performed for 4 different fractions /of core inventory/ released, various
meteorological conditions, and some other data. No emergency action was considered for this case.

Mentioned above calculations had conservative character /one leakage category, strictly defined meteorological conditions for each computer run/. The calculations were performed for model population corresponding to uniform density and three models of town. So far source term literature data has been used as an input for this calculations.

3. NEEDS AND EXPECTATIONS

National needs in the area of Computer Aided Safety Analysis include short-term needs related to validation and extension of SAR's for Zarnowiec NPP as well as long-term needs oriented on evaluation and improvement of WWER NPP's safety. The analyses include various tasks in the area of HCS thermohydraulic, fuel behaviour, containment behaviour, source term evaluation and off-site consequences. Two groups of problems may be distinguished:
- Analyses of accident and transients which do not lead to meltdown of the core
- Analyses of severe accidents, including evaluation of radiological consequences.
1. INTRODUCTION

The nuclear power program in Spain with eight power stations in operation and two more in construction as well as the different types of them: One graphite-gas, two BWR, five PWR Westinghouse type and one PWR german type demand a very large regulatory provisions and licencing procedures.

Since 1984 the Nuclear Safety Council (CSN) and the Industry and Energy Department promoted a more extensive research activity in the field of reactor safety. Since then, Spain has carried on research within the scope of a program that has continuously been updated.

To insure the success of this research program the Spanish organizations decided to cooperate by using the most efficient means. In this group several organizations have participated besides the Nuclear Safety Council (CSN), the National Laboratory dependent of the Industry and Energy Department, the Energy, Environmental and Technological research Centre, the Electric Companies coordinated by UNESA, the University and other nuclear companies.

The program was stablished by several projects. Some of them in multinational cooperation within the scope of the OECD and EPRI, others promoted by the spanish authorities to develop our safety technology. In these projects the organizations contribute with all their means to optimising finantial support.

This program is increasing continuously and it is being directed
towards the international aims in the present and in the future of the nuclear energy technology.

2. STRATEGY

Where we started with this program we had the following purposes:

- To ensure that the Nuclear Safety Council (CSN) has in our country people and tools to assess all these items related with the causes and effects in accident sequences.

- To promote further development of safety technology.

- To promote the growing of specialist groups.

In order to attain these objectives the following actions are required:

- To participate in the international projects related with loss of coolant and severe accidents.

- To integrate all the groups that in Spain were working in this area.

- To establish specialist groups in concrete items.

- To establish organizations that can coordinate the work of the severe groups.

Besides these actions the Nuclear Safety Council and the Industry and Energy Department recommended to the electricity companies and to the nuclear industry the research programmes of interest for their industrial activities and operations.

3. ORGANIZATIONS

Continuously there is a contact between the Nuclear Safety Council,
the Industry and Energy Department, the Energetic, Environmental and Technological Research Center, the Nuclear Industries and the Electric Companies to analyse the strategy, to decide the projects we are interested in carrying out, to establish the means needed to do it and to organize the team that ought to work in it.

In every project we have a Steering Committee with representation of all the nuclear organizations. Besides this steering committee every project have a Project Leader and several groups of work.

The organizations contribute to the project with people, tools and provisions.

4. ACTIVITIES

These Spanish organizations have two different types of activities:

1) The prevention of accidents has priority over the control or mitigation of their consequences, so we have here the following projects

- Intergranular stress crack corrosion in BWR pipe.
- Stress crack corrosion and intergranular attack in steam generator tubes.
- Embrittlement of vessels material.
- Non-destructive examination.

All these projects are national projects but we have collaboration with laboratories in other countries.

2) The loss of coolant or severe accident projects are international projects, as the following ones:

- TRANSRAP project
- LOFT project
- LACE project
- PHEBUS project

3) Besides these projects we have also a very intensive activity in probabi -
listic risk analyses with different objectives. Our power stations are doing the level one of his plants, there are working groups developing methodology and there are groups that are increasing their knowledge on this area.

The summary of results of all these activities is:

1) We have laboratories where we can experimentally study the growing of intergranular stress corrosion cracks that we have generated in it.

2) We also have the possibility of studying the embrittlement of vessel materials in our laboratories.

3) We are ready to use thermohydraulic codes to analyse the loss of coolant accident and we have used them in our plants.

4) We are ready to use methodology to perform PRA.

5. FUTURE PANORAMIC VIEW

In the frame of our technological strategy we will intend to increase our work on the behaviour of the aerosols and fission products in containment, man-machine interaction and severe accidents.

In the frame of our organization strategy we will go on integrating all our research program in one program and we will follow with the international cooperation.

All this effort is focused toward the accident management and the prevention of the accident.
SAFETY RELATED NUCLEAR RESEARCH IN SWEDEN

The Swedish Nuclear Power Inspectorate, SKI, is the regulatory body under the Nuclear Activities Act (1984). SKI's principal duties are to licence, to oversee and promote safety at the nuclear facilities, and to make sure that fissionable material is handled and stored in accordance with existing laws and agreements. Direct responsibility for the safety of the nuclear facilities lies, however, with the owners.

SKI is also responsible for initiating, directing and evaluating research in the field of nuclear safety. Funding, requested by SKI from the Government and the Parliament, is financed by fees from the nuclear power utilities. The research budget for 1987/88 is 48 million SEK (about 7 million USD).

In addition to the SKI research programme, there are other Swedish research programmes in the nuclear field which are in several respects closely concerned with nuclear safety. Thus, research programmes, similarly financed by fees paid by the utilities, are conducted by the National Board for Spent Nuclear Fuel (about 5 MSEK/year) and by the National Institute for Radiation Protection (about 5 MSEK/year).

The Swedish Nuclear Fuel and Waste Management Company, owned by the utilities and assigned to manage all the nuclear waste outside the nuclear power plants, conducts a major research and development programme, directly concerned with nuclear safety (about 70 MSEK/year).

Furthermore, there are research and development programmes continuously under way at the nuclear power utilities. The improvements generally aimed at, in terms of performance and availability, clearly in most cases go along with enhanced safety. In the recent years the utilities have, in addition, made considerable research and development efforts in conjunction with the current nuclear safety programme aimed at mitigating the consequences of severe accidents, e.g. the MITRA project (Mitigation of Reactor Accidents) at the Swedish State Power Board, and similar projects at the other utilities.

This paper will be focusing on the SKI research programme, which is in considerable part being conducted in cooperation with organizations in Sweden as well as abroad.
THE SKI RESEARCH PROGRAMME

General Objectives and Management

The SKI nuclear safety research programme has following major objectives, essentially in order of priority:

1. to provide working knowledge as required for SKI's supervisory and licensing duties
2. to maintain expertise and research capability for present and future needs
3. to support fundamental research with potential for safety improvements at the nuclear facilities (development of safety systems excepted as being the responsibility of the utilities)

The research programme is coordinated within the SKI Office of Regulation. Emphasis is placed on ensuring identification of all vital safety issues, and on obtaining balanced views on the safety for proper priorities to be made, by continuously reviewing the programme on a broadest possible basis, i.e. involving expertise also outside of SKI.

Thus, among other advisory committees tied to SKI, there is a Research Committee, composed of distinguished specialists in various fields, giving advice on the general direction of the research programme. Furthermore, for most of the particular domains covered below, Research Reference Groups have been appointed, composed of members from the utilities and various research organizations, partly on an international basis.

While thus providing for extensive expert review of the research programme, SKI retains executive power, in accordance with its obligation to provide independent safety assessments.

Emphasis is likewise placed on conducting the research programmes in international cooperation to the extent possible. Many examples will be pointed out below, under the various research domains. A significant part of the research is organized within the Nordic Nuclear Safety Research Programme (NKA), aimed at promoting nuclear safety on an internordic basis by getting eminent experts from the various countries together, and by reaching consensus in vital nuclear safety issues of common interest. The current 1985-1989 campaign of the NKA programme covers following issues:

- Release of radioactivity and consequential environmental effects
- Management of nuclear waste
- Risk analysis and the philosophy of safety
- Materials research - Corrosion and fracture mechanics
- The application of advanced information systems and expert systems in accident management
Human Factors

Objectives:
Assessment of the influence of work organization, working environment and conditions, education, basic and re-current training, and information systems on the safety of the nuclear plants in regard of the potentials and the limitations of the human factor.

Support level:
1985/86 1986/87 1987/88
1.2 MSEK (forecast) (budget)
1.5 MSEK 3.0 MSEK

The support is intended to remain at least at current level. The Swedish participation in the OECD Halden Reactor Project is in addition to the support stated above.

After the TMI-2 accident in 1979 the human factor issues were pointed out by the Swedish Government Committee on Nuclear Safety, set up for assessment of the impact of the accident, as being particularly important in its proposed programme for enhancing the safety of the Swedish nuclear plants. An increased emphasis is accordingly being placed on human factors research.

SKI sponsors the Swedish participation in the OECD Halden Reactor Project, an important part of which concerns research in the field of Man-Machine Communications, and in the Nordic Nuclear Safety Research Programme (NKA), covering, among several other subjects relevant to nuclear safety, important aspects of human factors such as control room design, reliability aspects of testing and maintenance and reliability aspects of work organization. More recently, advanced concepts in information technology have been addressed in the NKA programme, including technical as well as human factors aspects of Knowledge Based Systems and Expert Systems, primarily in regard of applications in accident management. Validation of advanced information technology also forms an essential part of the research programme conducted under the OECD Halden Reactor Project, covering in addition advanced aids for operator training (Computer Aided Instruction, CAI).

In conjunction with the Swedish participation in the international research project Programme for Inspection of Steel Components (PISC), SKI will contribute by sponsoring research concerning the influence of the human factor on the quality of non destructive testing (NDT). Such influence, partly related to cognitive aspects of the interaction of the operator with the testing equipment and partly to stress factors, is currently recognized as an important limitation in regard of achieving reliable fault detection. The project aims at improving the situation in realization of the vital importance of the role of the operator.

A comprehensive research programme has been initiated in cooperation between SKI and one of the Swedish utilities concerning procedures for validation the performance of control rooms and various comple-
mentary operator aids, usually computerized. Emphasis is being placed on the safety aspects in consideration of the variety of human factors having impact. The objective is to establish a basis for safety assessments of proposed measures for improvements in the control rooms such as, for example, alarm reduction systems, implementation of safety parameter display systems, success path monitoring systems, expert systems etc.

Other research projects concern methods of accounting for the human factor in probabilistic safety assessments (PSA), operator training schemes intended for individual training as well as group training, including training based on use of simulators, and problems related to operator alertness and performance on night-shift service.

Structural Integrity and Materials

Objectives: To develop the scientific and engineering basis for assessment of the structural integrity of nuclear power plant components of importance to safety

(1985/86) (forecast) (budget)
5.6 MSEK 9.4 MSEK 8.8 MSEK

Some further increase of the support level is anticipated

Research is under way, partly under the Nordic Nuclear Safety Research Programme (NKA), concerning linear as well as non-linear, quasi-static fracture mechanics, the conditions for rapid crack propagation vs. crack arrest to occur, and the initiation and propagation of cracks caused by thermal stresses.

SKI participates together with the Swedish utilities in the International Piping Integrity Research Group, IPIRG, organized by the USNRC. The general objective of IPIRG is to develop, improve and verify engineering methods for assessing the integrity of nuclear power plant piping with defects. The programme includes large scale piping fracture experiments.

Intergranular stress corrosion cracking (IGSCC) remains an important subject for research. In a comprehensive research programme, financed jointly by EPRI and SKI at the ASEA-ATOM laboratories since 1984, the influence of a range of common or potential impurities in BWR coolant water on the susceptibility of stainless steel to IGSCC is investigated. In another research programme, conducted by the utilities and partly supported by SKI, the prevention of IGSCC by addition of hydrogen to the water is being studied in actual BWR plant scale.

Irradiation assisted stress corrosion cracking of austenitic materials, IASCC, recently identified as a major problem for core components in LWR, is the subject of an extensive research programme.
jointly sponsored by the Swedish utilities and SKI. A comprehensive sample irradiation programme covering periods of up to 5 years, is under way in core positions in one of the Barsebäck nuclear power units. The programme aims at determining dose limits for sensitization of materials in reactor vessel internal structures, and exploring alternative materials, and water chemistry conditions, for improved resistance to IASCC.

Concerning ageing effects, research programmes are under way in regard of irradiation embrittlement of welds in reactor vessels and thermal ageing embrittlement of cast stainless steel.

Nondestructive materials testing is currently subject of extensive research, partly reflecting an urgent need for improvements of the available testing procedures as well as the growing need for testing with regard to the ageing of the nuclear power plants. Current emphasis is on ultrasonic testing.

SKI and the Swedish utilities participate in the international Programme for Inspection of Steel Components (PISC), the third part of which, presently under way, concerns the reliability of the testing procedures when applied for the detection of cracks developed in service. SKI also participates in a Nordic programme, NORDTEST, for assessment and development of nondestructive testing methods.

In addition research is under way, contracted by SKI, for modeling of ultrasonic beam propagation, as an aid in assessment of ultrasonic testing strategies, and for validation of methods which have been developed for sizing of detected faults and cracks.

Strategies for optimal planning of recurrent in-plant component testing programmes remain an important topic in the SKI materials research programme.

**Thermohydraulics**

**Objectives:** Methods for assessing the safety margins, mainly in regard of maintaining, at all times, proper cooling of the reactor core, considering possible transients and loss of coolant accidents

**Support level:**

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<th>Year</th>
<th>1985/86</th>
<th>1986/87 (forecast)</th>
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<tr>
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<td>6.9 SEK</td>
<td>6.5 MSEK</td>
<td>5.4 MSEK</td>
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The support will be settling on a reduced level following the completion of most major experimental programmes required for code developments and validations.

A general aim, in regard of the use of thermohydraulic computer codes in licensing, is to develop methods of safety assessments based on best estimate calculations and uncertainty estimations to replace former methods based on conservative modeling. Work in this area is
under way at the ASEA-ATOM and Studsvik research organizations under research contracts with SKI. SKI thus participates in the International Code Assessment Program (ICAP) aiming at quantitative uncertainty assessment of the thermohydraulic computer codes, including RELAP5 and TRAC-F, and recommendations for the modeling of systems and phenomena. The Swedish contribution to the present ICAP Assessment Matrix, covering 10 thermohydraulic experiments, has been completed.

Continuing support is granted for validation of the thermohydraulic code, GOBLIN, and the transient analysis code BISON, both developed by ASEA-ATOM and used for licensing purposes concerning BWR. The basis of current validation work is partly experimental results available from previous programmes in Sweden and abroad, including participation in international standard problems (ISP), and partly benchmarking against RELAP5 in actual LWR LOCA and transient analyses.

A study is being conducted to explore the possibility of benchmarking the severe accident analysis code MAAP (Modular Accident Analysis Program) against GOBLIN. The aim is to help verifying the MAAP code for analysis of severe accidents with recovery, involving temporary uncoverage of the reactor core.

The needs for experiments in thermohydraulics, strongly supported by SKI in the past, appear at present to be largely exhausted. A major experimental programme concerning emergency cooling of the ASEA-ATOM SVEA fuel assembly has recently been successfully completed, demonstrating the predicted improvement of the coolability of this type of fuel under emergency cooling conditions.

Under a SKI research grant, an effort has been made to compile various results of thermohydraulic analyses of the behaviour of the nuclear plant systems under transient conditions in a suitable form for reference and training purposes. The compilation has been made available as a 'Handbook in Relations between Process Parameters under Transient Conditions in Swedish BWR's'. A similar effort is underway for PWR's.

**Nuclear Fuels**

**Objectives:**

Methods for assessing the safety margins in regard of the integrity of the reactor fuel under design basis conditions, including transients and loss of coolant accidents, with particular regard to the ability of the fuel to retain the fission products

**Support level:**

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<td>1.9 MSEK</td>
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<td>1987/88</td>
<td>2.5 MSEK</td>
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The Swedish participation in the OECD Halden Reactor Programme is in addition to this support.
SKI participates in a number of international experimental programmes for investigation of various phenomena of interest in regard of the ability of reactor fuel to remain adequately intact and to retain the fission products under conditions of high burn-up and power transients. Such programmes include the OECD Halden Reactor Project, the already completed Risø Transient Fission Gas Projects I and II, the Battelle 'High Burnup Effects Program' and the Belgonucléare/CEEN TRIBULATION project.

SKI also participates in international fuel test programmes operated by the Studsvik research organization in the R2 research reactor. These include the RAMP programmes, largely concerning pellet clad interaction failures (PCI) and the ROPE programme (Rod OverPressure Experiment), concerned with fuel rod and cladding behaviour at excess end-of-life internal vs. external pressure.

Studies of the behaviour of a fuel rods at various power levels are being conducted in Studsvik, jointly financed by SKI and the utilities. Novel means of determining the thermal response characteristics of the fuel will be used, based on noise analysis using artificially superimposed power noise. Improved ceramographic methods will be utilized for the characterization of the fuel in the post irradiation examinations.

Research contracted at ASEA-ATOM for improved modeling and computer codes for the estimation of fission product release from fuel, accounting properly for the irradiation history, has reached an advanced state of progress.

**Nuclear Materials Safety and Safeguards**

**Objectives:** Development of knowledge required for assessing the safety and safeguards aspects of transport, handling and storage of fissile materials

**Support level:**

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<td>1.6</td>
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The research programme includes participation in the Nordic Nuclear Safety Research Programme (NKA), which covers in parts certain aspects in this field. Thus, one of the NKA projects is concerned with methods for analysing the needs for transportation of fuel and various types of waste products in conjunction with normal operation, maintenance, and decommissioning of reactor power plants. In another project the quality assurance aspects of transportation containers are being analysed with a view to establish guidelines for design, manufacturing and periodic inspection.

A research programme is under way aiming at verifying the computer codes for criticality and heat transfer calculations used for safety assessment of transport containers for fissile materials.
Since completion of the INTERTRAN code for estimation of radiological risks associated with transport of radioactive materials, application support is presently provided under a SKI research grant.

Non-destructive methods for estimation of the contents of fissile materials in irradiated nuclear fuel elements are being developed and validated in cooperation between SKI and the Finnish Centre for Radiation and Nuclear Safety. The method evaluated in Sweden is based on using the gamma emitting isotope Cs-137 as an indicator.

In another research programme the uncertainties in estimating MUF (Material Unaccounted For) are being evaluated with a view to finding ways of improving the accuracy of the accounting systems.

**Severe Accidents and Plant Internal Accident Management**

Objectives: Knowledge base and methods for estimating source terms in the event of severe nuclear accidents in Swedish LWR:s, and for performance assessment of mitigating systems and procedures.

Support level: 

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<th>Year</th>
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<td>3.7 MSEK</td>
<td>2.3 MSEK</td>
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The support is gradually diminished as the severe accident mitigation programme is now approaching completion according to plan (1988).

The research programme in the field of severe reactor accidents is largely governed by the severe accident mitigation programme enforced by the Swedish government in 1981 on the basis of a review of the nuclear safety issues undertaken by a Swedish Government Committee on Nuclear Safety because of the accident in TMI-2. The programme provides guidelines on prevention and mitigation of radioactive releases in case of severe reactor accidents. Time frames were fixed for implementation of the proposed mitigative systems: the FILTRA system for filtered containment venting in the two Barsebäck BWR units to be completed by 1985 and corresponding mitigations systems to be implemented on the remaining 10 units by 1988.

Accordingly, the initial research programme was tied to the FILTRA project. The following RAMA project (Reactor Accident Mitigation Analysis) has been aiming at providing for the research needs associated with the remaining three PWR:s and seven BWR:s.

The main task of the RAMA project has been to develop suitable analytical tools - computer codes - required for site specific safety analyses, and to validate these tools. The MAAP code, developed by IDCOR (the US Industry Degraded Core Rulemaking program), was chosen as the main tool and the RAMA project accordingly joined IDCOR.
The RAMA project was split in phases, of which the second phase, RAMA II, will be completed by mid 1987. Particular objectives of this second phase were to implement a new version of the MAAP code, which had been developed by IDCOR, and to develop plant specific adaptions of the code, ready for practical use by the utilities. Another aim was to consolidate the scientific basis of the MAAP code and of the general conclusions of the RAMA project. As important research on the international scene is still in progress the matter of scientific consolidation will again receive attention in the third and last phase, RAMA III to be completed by mid 1989.

The main subjects of the RAMA III project are as follows:

- Comparison of the MAAP code with the corresponding USNRC risk analysis codes STCP (Source Term Code Package) and MELCOR
- Complementary research on certain important severe accident phenomenology such as core melt coolability and certain questions related to chemical behaviour.
- Validation schemes for site specific, internal accident management procedures. Development of computerized training aids based on simulation
- Handbook documenting the present basis for estimating source terms from severe reactor accidents and for assessing the performance and the safety of mitigative systems and procedures

International cooperation forms an essential part in the RAMA project, participating in the following programmes:

- The Severe Fuel Damage, Containment Loads, and Source Term Research Program (USNRC)
- The EPRI LACE program (LWR Aerosol Containment Experiment)
- The Marviken MX-V-ATT Aerosol Transport Tests operated by Studsvik Energiteknik, Sweden
- The OECD LOFT project operated by EG&G, particularly the fission products tests, FP-1 and FP-2
- The CSNI Joint Task Group on TMI-2
- The Nordic Safety Research Programme (NKA), one part of which covers activity release and consequential environmental effects caused by severe reactor accidents

The results of the RAMA project have been serving the needs of the utilities in their development of the mitigative systems all since completion of the first phase of the RAMA project in the beginning of 1985. The results of the final phases will constitute an important basis for making the safety assessments required in issuing permits for continued operation of ten Swedish nuclear power plants beyond 1988.
Systematic Safety Analysis

Objectives: Methods for systematic safety and reliability analysis of nuclear power systems employed for periodic safety assessments of the Swedish nuclear power plants

Support level:

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<th>Year</th>
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<td>1985/86</td>
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<td>1987/88</td>
<td>3.6 MSEK (budget)</td>
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Advanced methods and computerized tools are presently available and extensively used at the Swedish utilities in the ongoing, periodic safety analysis programme requiring the utilities to prepare ASARs for each plant (As operated Safety Analysis Report). Such analyses, including 'level 1' probabilistic safety assessments (i.e. terminating the set of events at the point of core damage), in most cases limited to internally initiated events, will be completed for all plants in a first round by 1990.

Current research programmes are aimed at complementary developments and improvements of the methods in certain important respects.

An important part of current research is tied to the Nordic Safety Research Programme (NKA) which includes one project aimed at study of the role of the technical regulations, imposed upon the operation of the nuclear power plants, to determine the actual safety implications of the regulations, positive or negative, and the possibly available scope for optimization. In another project the significance of modeling completeness and various modeling assumptions and limitations are being investigated by means of benchmarking exercises.

A major research programme, named SUPER-ASAR, was recently launched, aiming at comprehensive comparative analysis of already performed ASARs in order to evaluate the methodology on the basis of differences that can be observed between results obtained as correlated to different modeling, incompletenesses and other factors. A basis for making priorities in this field of research should also be obtained.

Further research programmes concern:

- Sensitivity analyses and uncertainty analyses: development of methods employing distribution functions rather than average reliability data
- Methods for calculating risk indexes for classifying faults with respect to their safety significance
- Principles for the structuring of reliability data bases, including reported failure data and data concerning common cause failures (CCF)
- Methods for safety analysis with respect to external events, like fire, flooding, seismic events etc
Methods for trend analysis for detection of gradual deterioration of systems and components

Accounting for the human factor in PSA

Special Safety Issues

Objectives: Knowledge and methods for evaluating certain risk factors, e.g. seismic risks. Evaluation of risk associated with nuclear power production abroad

(1985/86) (forecast) (budget)
0.6 MSEK 1.9 MSEK 2.0 MSEK

While, compared with many other countries, Sweden is well removed from the so-called plate boundaries, constituting sites for most of the world's larger earthquakes, and accordingly experiences low seismic activity, seismic risks cannot be totally excluded.

A major research project, jointly financed by SKI and the National Board for Spent Nuclear Fuel, was carried out by the National Defence Research Institute during the years 1980-1984 in an attempt to provide an improved basis for assessing the seismic risks in southern Sweden. A digital dense seismic network was established and detailed source information was obtained from about 250 earthquakes of magnitude 4.5 and less. The study provided confirmation of previous estimates of the seismic risks.

The continued research has been concerned with modeling of the propagation of seismic waves for computation of the ground motions to be expected at the sites of the nuclear power plants as a result of any given earthquake. Furthermore, in order to provide required input data in the models, complementary geotectonic surveys of the regions surrounding the sites are presently being conducted. Probabilistic risk assessments will follow on the basis thus established, to be completed for the Barsebäck and Ringhals plants about mid 1988. This project constitutes a joint effort by SKI and the utilities involving, among others, the Seismological Department of the Uppsala University, the Department of Geology at the University of Lund and the Swedish Geological Survey.

As a result of the Tjernobyl accident, which required a great deal of expert involvement for assessing the safety implications in Sweden, research funds have been allocated for activities aimed at developing certain knowledge bases for reference in similar emergency situations.
Emergency and Early Warning Monitoring Systems

Objectives: Methods for reliability assessment of components in safety related monitoring systems. Assessment of needs and techniques for early warning systems.

Support level:

<table>
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<th>Year</th>
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<th>1986/87</th>
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<tbody>
<tr>
<td>(forecast)</td>
<td>0.8 MSEK</td>
<td>0.6 MSEK</td>
<td>1.4 MSEK</td>
</tr>
</tbody>
</table>

SKI and the utilities conduct jointly a comprehensive research programme to provide improved scientific basis for guidelines and regulations in regard of qualifications needed for safety related components intended for long term service in the reactor containments. The programme covers cables, O-rings, connectors, relays, temperature sensors, microswitches, magnetic valves etc, and validation of methods for long term performance qualification for various environmental conditions based on accelerated testing and so-called ongoing qualification programmes.

Research projects at the Studsvik research organization have previously been supported concerning applications of noise analysis in monitoring of important state parameters in nuclear power stations under emergency conditions with restricted possibilities of access. Current research activities are directed towards possible applications in early warning systems, employing spectral recognition or process parameter identification methods using computer models of the processes being monitored.

The safety aspects of using of computer hardware and software in safety related systems are receiving increasing attention in the research programme.

Nuclear Waste

Objectives: Knowledge base and methods for safety assessments with regard to handling, storage, transport, and final disposal of nuclear waste.

Support level:

<table>
<thead>
<tr>
<th>Year</th>
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<td>6.8 MSEK</td>
<td>8.1 MSEK</td>
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</table>

The Swedish plans for nuclear waste management foresee a final repository deep in the bedrock to be operational by the year 2020. An extensive research and development programme is under way to this end, conducted by the utility owned Swedish Nuclear Fuel and Waste Management Company (SKB).

SKI conducts an independent research programme in preparation for assessments of the safety reports expected at various stages of the development and implementation of the final repository.
The SKI research programme is conducted in extensive international cooperation. A leading role has been taken by SKI in a number of international code verification and validation projects, including INTRACOIN (radionuclide migration in rock; completed), HYDROCOIN (groundwater flow; partly completed), and the recently launched INTRAVAL (geosphere transport models). Parts of the programme are conducted under the Nordic Nuclear Safety Research Programme (NKA).

In addition to the international projects research is under way to explore a number of issues important for safety assessments of final repositories, including uncertainty analysis of hydrology predictions, three-dimensional modeling of radionuclide transport, transport modeling of fractured rock, and various problems concerning the interactions of the repository with surrounding rock. Another important issue relates to application of probabilistic techniques for safety assessments of final repositories.

Furthermore, SKI supports fundamental studies in geology and geochemistry aimed at consolidating the scientific basis in the modeling of radionuclide migration and evaluating the stability of engineered radioactivity barriers.

The SKI research activities aimed at enabling preliminary safety assessments of the final repository, to be made in the beginning of the nineties, are coordinated under the heading PROJECT-90. The Studsvik research organization, the Stockholm based KEMAKTA consulting company, and a number of universities and institutes, partly foreign, participate in the programme, guided by an international team of experts.

Research concerning nuclear waste handling in conjunction with reactor accidents and the final decommissioning of nuclear power plants is conducted within the Nordic Nuclear Safety Research Programme (NKA). The aims are to identify problem areas and to propose suitable technical and procedural solutions as well as preliminary guidelines for safety assurance. In the project dealing with radioactivity release from an accident the release to the reactor containment is assumed to amount to 10% of the core inventory of volatile fission products.

A project concerning decontamination of a PWR steam generator, and related safety aspects, was recently completed with partial support from the SKI in a joint research effort by the Swedish State Power Board, Studsvik Energiteknik, and ASEAA-ATOM. The procedure intended for use in a projected exchange of steam generators in the Ringhals nuclear power station, was evaluated and found to enable significant dose reduction to be effected while associated safety concerns and waste problems were found manageable.
General support of nuclear safety research

              (forecast)  (budget)

7.7 MSEK      9.0 MSEK      10.0 MSEK

The support referred to under this section includes the Swedish contribution to the research fund of the Nordic Council of Ministers, from which grants are then allocated to projects in the Nordic Nuclear Safety Research Programme (NKA) on the condition of at least equivalent financing being obtained from the national organizations, and to the OECD Halden Reactor Project.

SKI also sponsors a chair in Nuclear Safety at the Royal Institute of Technology. Furthermore, SKI sponsors safety committees in the countries where nuclear power stations are located. This latter support is related to research in regard of the obligation laid upon SKI to inform the public about nuclear safety, including the results achieved in the research programme.
Brief Report on activities in Sweden in the area of source term etc.

General

In February 1986, the Swedish Government decided on the conditions for operating licences to be granted for operation beyond 1987 of the 10 reactors in Sweden to be provided with additional measures for mitigating radioactive releases in the event of a severe accident. Such measures were previously implemented at the Barsebäck plant (FILTRA).

The Inspectorate has contracted the Gesellschaft fuer Reaktorsicherheit (GRS) to assist in the ongoing safety review work.

Preliminary safety reports on the proposed measures, consisting of additional systems, system modifications and other, were submitted to the Inspectorate in November 1986. Permissions from the Inspectorate for implementation is required whenever expected to effect the safety. Final safety reports will be submitted prior to final completion and permits for continued operation after 1988 will be granted by the Inspectorate on fulfillment of all pertinent conditions.

Implications of Chernobyl

The safety implications of the Chernobyl accident were assessed in a report by an Expert Group set up by the Minister for Industry ("After Chernobyl - Consequences for energy policy, nuclear safety, radiation and environmental protection". Swedish Ministry of Industry, Dal 1986:11, English edition). It was concluded that the direction of the ongoing programme for improved measures for protection of the environment from nuclear accidents was in no need of modification, but that the urgency of the program had been illustrated. Accordingly, measures have been taken to speed up the program.

The Chernobyl accident gave rise to an increased concern about residual risks, i.e. the highly improbable events remaining uncovered by currently foreseen safety measures in the nuclear power plants. Specialist seminars were therefore conducted for review of the safety and the risks associated with the integrity of reactor vessels, seismic events, and possible uncertainties in present safety assessments, e.g. in regard of reactivity events and human factors. The conclusions reached at these seminars were part of the basis for the conclusion reached by the Expert group, as referred to above.
Research

The RAMA II research project (Reactor Accident Mitigation Analysis), conducted for the purpose of improving and validating the general knowledge base and the computer codes (mainly the MAAP code) used for the site specific analyses, is due for completion according to plan in mid 1987. Plant specific adoption of the MAAP code to all the Swedish nuclear power plants is one fundamental achievement of the RAMA II project.

A further validation effort will be made in a follow-on project, RAMA III, to be completed by 1988/1989. The RAMA projects are financed jointly by the Inspectorate, the Radiation protection Institute and the utilities. The total research efforts have passed the maximum and are currently gradually reduced.

Research activities in the RAMA III project include, as main parts, a code comparison program aimed at validation of the MAAP code by comparison with USNRC codes (STCP, MELCOR), and a program for reviewing certain accident management schemes.

Through the RAMA projects, continued Swedish participation is provided for in the international projects SFD (USNRC), LACE (EPRI), OECD-LOFT (FP-2) and the TMI-2 research programmes (NEA/CSNI).

Status of the mitigative measures implementation programme

The Forsmark 1, 2, and 3 units and Oskarshamn, unit 3, have been all provided with means of flooding the pedestal area with water in case of an accident with imminent vessel melt-through and instructions for performing such procedure have been prepared.

Barriers for protecting containment penetrations against the core melt are being designed and a test program is under way for assessing their performance.

Design work is under way on provisions to supply the containment spray system from the fire protection system or by mobile pumps, on the containment pressure relief systems and on systems for emergency monitoring systems.

The utilities have decided on the choice of filter system for filtered venting of the containments, the ASEA-ATOM FILTRA MVSS, a multiventuri filter system for removal of aerosols and elementary iodine.

The development of emergency operating procedures, in progress at each one of the utilities, has reached an advanced stage and is expected to enable validations to commence by mid 1987 followed by training of the personnel scheduled for completion in the autumn of 1988.
Research Activities in Switzerland
compiled by:
D. Haschke, O. Mercier
Swiss Federal Institute for Reactor Research
CH-5303 Würenlingen, Switzerland

Introduction:

Thermal reactor safety research in Switzerland is done for presently operating light water reactors, consisting both of BWR's and PWR's of American and German Origin. Safety research is addressing two main areas:

- Operational safety and prevention of severe accidents.
- Severe accidents and source term research.

Operational safety and prevention of severe accidents (LWR-accidents):

- OECD-LOFT participation: LOFT-FP-2 post-test thermal hydraulic and fission product analyses
- PHDR-participation: structural analysis (seismic tests), crack growth analysis (thermoschock tests) and crack detection (acoustic emission)
- Radiation embrittlement of reactor vessel steel (IWG-RRCP and HSST programs): simulation of the embrittlement of various steels by accelerated irradiation in a test reactor (SAPHIR) and measurement of the change of their mechanical and physical properties.
Outlook: Shifting emphasis on calculation of safety margins, associated with degradation of properties of primary circuit components.

- Embrittlement and corrosion of pressure vessel steels, piping steels, etc. Contamination of piping.

- Non-destructive testing of components (acoustic emission, ultrasonic inspection)

- Fast simulation of transients and accidental conditions. Accident management.

Severe Accidents and Source Term Research:

Source Term Research in Switzerland has been experimentally oriented during the last few years and centered mainly around activities associated with fission product and aerosol behaviour during severe accidents with core melt. In particular, the following lines of activities have been followed:

Iodine Chemistry:

Prime concern was the iodine chemistry in aqueous solution and partition between aqueous and gaseous phase. Experiments have been performed to measure iodine partition coefficients in clean conditions and as a function of pH, Redox-Potential and temperature. These experiments were expanded to add both Cs and Ag in the aqueous phase to include their effect on iodine partition and in a last step the effect of a strong Gamma radiation on the Cs-Ag-I system in aqueous solution was investigated.

EIR has participated in two large scale aerosol experiments, DEMONA and LACE, which are completed now. For DEMONA, which was executed as a German - Swiss cooperative effort, EIR developed an aerosol measurement and data acquisition system which was capable of measuring both aerosol mass concentration and size distribution as well as water droplet size distribution in post accident containment atmosphere of about 130 °C, 0.3 MPa total pressure and water vapor to air ratios of 0.5 to 1.0. EIR was also responsible for the operation of the measurement system during the DEMONA tests and for aerosol data analysis.

For the international LACE project, which was carried out as an international effort with about 10 countries and/or institutions participating, EIR supplied special measuring techniques (transmissiometers and droplet spectrometers) and operated them during three tests.

Presently, EIR has three pilot-scale experiments in progress addressing the following problems:

- resuspension of fission products out of an evaporating sump
- resuspension of aerosols deposited on containment surfaces ("PARESS")
- iodine and aerosol mass transfer in a scrubbing pool ("POSEIDON")

In addition, EIR contributed to the formulation and is helping in the formation of the international advanced containment experiment ACE, which will address such areas as evaluation of various containment emergency filtering techniques and strategies, transport and removal of CsI, HI and CH$_3$I - I$_2$ both under basic pH conditions and with simultaneous H$_2$-burning and certain aspects fission product release and molten core concrete interaction with simultaneous water flooding.

Furthermore, EIR is now in the preparatory stage to start individual source term analysis for Swiss Nuclear Power plants starting with a BWR-3 reactor with a Mark-I containment.
PRESENT STATUS OF NUCLEAR RESEARCH ACTIVITIES IN TURKEY

T.TURKER

PREFACE

To the Technical Committee Meeting on Thermal Reactor Safety Research from 8-11 Jun. 1987 at the International Atomic Energy Agency's Headquarters in Vienna we will present the following three projects. Two of them are supported by the Agency, and the last one is a joint project.

1- A Case Study for TR-2 Reactor

This project is a Coordinated Research Programme (CRP) on the "Probabilistic Safety Assessment for Research Reactors" and supported by International Atomic Energy Agency (IAEA). The overall objectives of the project are:

- Systematic review of TR-2 research reactor system
- Determination of accident scenarios for TR-2
- Data base development by using the experience and data collected from TR-1 reactor which is already decommissioned
- Related to this data base Human Error modelling and development.

There are ten countries participating in this Coordinated Research Programme.

2- Turkish Akkuyu Probabilistic Safety Evaluation (TAPSE) Project

TAPSE project was started in Jan. 1986 with support of International Atomic Agency. The overall objectives of the
The project are

- To develop expertise in Probabilistic Safety Evaluation (PSE) and particularly Safety Design Matrices (SDMs) in Turkey in order to assist the licensing of Akkuyu-I a CANDU 600 NPP.

3- Core Conversion

Core Conversion is a joint project between Çekmece Nucl. Res. Center-Turkey and ANL-USA. The primary objective of the project is to replace currently used highly enriched uranium (93% U-235) fuel in TR-2 reactor, with low enriched uranium (20% U-235) fuel, without significantly degrading the performance of TR-1 and TR-2 research reactors. According to this main objectives, the project covers all the neutronic and safety related analysis.
I- INTRODUCTION

The status of nuclear research activities in Turkey related to thermal reactor safety, can be summarized as:

1. Case study for TR-2 research reactor
2. Turkish Akkuyu probabilistic safety evaluation (TAPSE) project
3. Core conversion (HERTR) for Turkish research reactors.

In the following paragraph, for each activity above, brief and summary descriptions are given.

II- SUMMARY DESCRIPTIONS OF RESEARCH ACTIVITIES

1- Case Study For TR-2 Research Reactor.

The main idea of carrying out Probabilistic Safety Assessment (PSA) for the TR-2 research reactor was emerged from the following research needs:

- The full review of the TR-2 system and procedures from the safety point of view.
- The accumulation of knowledge and gaining expertise on PSA applications for future uses in power reactor safety.
The understanding and modelling of human behaviour in safety related matters under the conditions of country.

The research contract proposal for the IAEA Coordinated research Programme (CRP) on the "Probabilistic Safety Assessment for Research Reactors" was prepared in January 1986 and the first year of the research contract namely "Case Study for TR-2 Reactor" was commenced in April 1986 under the contract no: 4383/RE.

There are ten countries participating in CRP (Argentina, Australia, Austria, Czechoslovakia, Peru, Switzerland, UK, USA, and Turkey) and the whole project will continue for about 3 years.

Within the framework of the general scope of the CRP set by the IAEA, our main objectives are as follows:

- Systematic review of TR-2 systems.
- Determination of accident scenarios for TR-2.
- Database development by using the experience and data collected from TR-2 reactor which is already decommissioned.
- Related to this database, human error modelling and development.

The work plan for the first year is outlined below:

1- Study of the TR-2 reactor systems
   - Obtain detailed design descriptions from design documents
   - Draw cooling system flow diagrams
   - Draw control and instrumentation flow diagrams.
2- Start to establish a component labeling scheme.
3- Start development of the database.
4- Define qualitatively initiating events (including externals).
5- Start to work on fault tree construction of the
Since the beginning of the research, most of the effort is spent for assembling a large amount of information.

The basic characteristics of TR-2 reactor are as follows:

It is a 5 MW thermal power, swimming pool type research reactor moderated and cooled by light water. Present core consist of 19 standard fuel elements, 4 control elements and 6 beryllium elements as a reflector.

Fuel elements are TR type containing 95% enriched U-235 in a U-Al alloy cladded with Al. Each fuel element has 23 plates contains 280 g U-235. Each control element has 17 fuel plates having channels for sliding of control rod and contain 208 g U-235.

Control rods are fork type double blades Ag-In-Cd alloy suspended by electromagnets.

Initiating Event Description:

For the development of accident scenarios it is first necessary to identify and define events that could initiate an accident. These events could be internal to the reactor system or external of the system. Internal events occur due to system failure or malfunctions, component failures or human errors.

The identification of the initiating event for the TR-2 accident sequences is made through the use of:

1) The generic data for initiating events

The categories of initiating events that may be similar in each research reactor are identified in the IAEA reference document. The following categories of initiating events are thought to be applicable to TR-2 reactor:

- Loss of electrical power supply
- Loss of coolant accident
- Insertion of excess reactivity
- Erroneous handling or failure of equipment or component

2) The determination of the system-specific initiating events

The TR-2 log books are being examined for this purpose.

All mitigating systems and human actions considered for developing accident scenarios for TR-2 are listed below:

Scram System
  automatic-mechanic
  manual (human action)

Electric power system
  ac power
  dc power

Emergency power system
  diesel generator
  stable generator

Reactor cooling system
  forced circulation
  natural convection

Reactor building air circulation system
  ventilation fans
  emergency ventilation

Iodine removal system (filters).
Accident Scenarios:

Systematic event trees are developed qualitatively for a certain group of initiating events in order to delineate the accident sequences to be considered in the analysis.

These are:

LOCA

Loss of flow accident, forced circulation unavailable (flow blockage)

Excess reactivity insertion

Excess reactivity leading to super prompt critical conditions

LOSP (Loss of Electrical Power Supply)

Event trees are shown in Figs., 1-6.

For developing and displaying all possible accident sequences started with the above mentioned initiating events, the information developed in the system familiarization stage is used. All front line systems (systems that must prevent accident or mitigate consequences), normal systems, support systems and operator actions needed to mitigate or terminate the progression of events are considered and addressed whenever necessary. However, only the front line systems and operator actions appear on event trees.

For example, in LOCA-event tree, "human intervention" which appears as heading on the tree is included for the operator action manually tripping of the forced circulation pump and closing isolation valves. This action is considered necessary for reduction of the effect of LOCA and prevention of further coolant losses.

Once the initiating event and front line systems are identified, the set of possible failure or success states for each
front line systems are defined. All success-failure states combined together to obtain several accident sequences for a given initiating event.

The determination of the state of the systems and the possibility of activity release at the end of each event sequence is decided only qualitatively at present. However, it is necessary to undertake transient and thermohydraulics analysis and fault tree analysis in order to define success-failure states and the frequency of these states quantitatively. This quantitative evaluation is not within the scope of the first year of the project and will be undertaken during the next two years as the work progresses.

Human Reliability Analysis:

The initiation of the human reliability analysis and determination of the human errors is one of the major tasks of the project. For this purpose, in parallel to the system analysis, research has been initiated in cooperation with a group (psychologist) from the Industrial Health Section of the University of Istanbul. The preliminary work is described briefly below:

The working conditions in reactor control room is examined. Then "work analysis" is carried out in order to specify conditions of the reactor operation and certain psychological tests have been evaluated for application to each person involved in operation and maintenance. The tests will be carried out by simulating abnormal conditions in the reactor control room in order to examine human behaviour under stress as well as normal operation.
Fig. 1 - Event Tree - LOCA
Fig. 2 - Event Tree - Loss of flow accident (forced circulation unavailable)
<table>
<thead>
<tr>
<th>Event Tree - Flow Blockage</th>
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</thead>
<tbody>
<tr>
<td>1 no activity release</td>
</tr>
<tr>
<td>2 activity release</td>
</tr>
<tr>
<td>3 activity release</td>
</tr>
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<td>4 activity release</td>
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<tr>
<td>13 activity release</td>
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<tr>
<td>14 activity release</td>
</tr>
<tr>
<td>15 large activity release</td>
</tr>
<tr>
<td>16 no activity release</td>
</tr>
<tr>
<td>17 activity release</td>
</tr>
</tbody>
</table>
Fig. 4 - Event Tree - Excess reactivity insertion
Fig. 5 - Event Tree - Excess reactivity leading to super prompt critical conditions
Loss of Off-Site Power

Automatic SCM

Manual SCM

Natural Circulation

Emergency Power

R.B. Air Circulation

Iodine Removal

1. No activity release
2. No activity release
3. No activity release
4. No activity release
5. Large activity release
6. Activity release
7. Activity release
8. Activity release
9. Activity release

Fig. 6 - Event Tree - LOSP (Loss of Electrical Power Supply)
2- Turkish Akkuyu Probabilistic Safety Evaluation (TAPSE) Project.

The TAPSE project was planned in June 1985 and started in Jan. 1986 with support of IAEA. The overall objectives of the project are:

- to develop expertise in PSE and particularly Safety Design Matrices (SDMs) in Turkey in order to assist the licensing of Akkuyu I, a CANDU 600 HPP.
- to provide a decision tool for safety and operational use for the lifetime of the plant.

The scope is to perform a level-1 PSE. It was not intended to finish a complete PSE of the plant during the time period reserved for the study. However, all tasks such as Event and Fault Tree Development, External Event Analysis, Human Reliability Analysis, Data Base Development, Uncertainty Analysis, Accident Sequence Quantification, etc. will be covered at certain extent.

Originally, the working schedule of the PSE study is intended to be valid for any type of reactor.

Therefore if the reactor type ordered by Turkey is changed, the prepared and ongoing programme and its schedule will be adopted to the new reactor system by utilizing the expertise already gained for the application of similar techniques with appropriate design data.

A final report has been published in Feb. 1987. The final report is the second large-scale information product of TAPSE project and reproduces the results obtained since Jan. 1986.
It covers the analysis of events started with an initiating event, "Loss of Total Feedwater with Class IV Power available", for a fictive CANDU plant. This problem is originated during the "PSE Workshop", held at Çakmecè Nucl.Res.Center-Istanbul, between Jan. 20-31, 1986, because of difficulties for obtaining the informations on a real CANDU 600 plant. The TAPSE team was asked to extend the analysis, that is Event/Fault Tree analysis, in time to one month in order to be sure that an adequate time for the repair of the main feedwater pumps would be provided.

During the 12 months passed, course documents provided by the Canadian experts during the workshop for a simplified CANDU plant and the summary safety reports submitted to the licencing authority of Turkey by AECL during the bidding negotiations of Akkuyu-1 NPP were only sources of information at hand both for analysis and quantification. The data base used in Event/Fault Tree analysis is basically AECL set provided during the workshop also.

Sometimes safety reports mentioned above were used to gather more information about the systems or functions, etc. for learning purposes or whenever a need had arisen without thinking the differences between two plants, i.e., fictive plant given during workshop and standard CANDU 600 NPP.

The Workshop example which analyzed the behavior of the plant for a short time (i.e., up to 30 minutes) after the initiation of the accident is extended to long term, i.e., up to one month. A summary of the results of this extension is given below. The initiating event is "Loss of Feedwater with Class IV Power available".
The frequency of initiating event has been estimated as 7.605×10^{-4} per year. This new value of frequency is about an order of magnitude lower than the one given during Workshop.

The availabilities of the following mitigating systems

- Shutdown cooling systems
- Condenser Steam Discharge Valves
- Condenser Cooling Water Supply System
- Auxiliary Feedwater System
- Pumped Emergency Water Supply System

have been obtained for long term operation by using data available in Workshop documents.

The screening of event tree developed for the initiating event, with a frequency of 3.305×10^{-4} per year has provided 43 accident sequences with frequency larger than 10^{-7} per year. Only two of the sequences have end points which require moderator as heat sink, otherwise plant is always stable with the help of mitigating systems.

The Event/Fault Tree and Event Sequence analysis applied to fictive CALDU plant for one single initiating event produced valuable information and is proved to be useful for gaining the methodology of PSA technique. The work done so far is general in nature and it is applicable to any type of reactor.

3. Core Conversion (REGET) for Turkish Research Reactors.

The objective of the Core Conversion project is to develop the technology to permit research and test reactors to convert from the use of fuels containing highly enriched uranium to the use of fuels containing low enriched (20% U-235) uranium without signifi-
cient penalties in core performance of fuel cycle cost, and without reducing the overall safety of the reactors.

Because of the advantage of high density fuels, Turkey has decided to use high density silicide fuels in the existing research reactors. TR-2 a pool type research reactor in 5 MW power is using presently 93% enriched HEU plate type fuel elements. It is concluded that without any change in fuel element geometry and no penalty in the present high enriched uranium fuel cycle burn-up performance, conversion to low enriched uranium (LEU) would be feasible within the limits of current fully qualified silicide fuel materials technology. There would be no significant adverse effect on safety related parameters (e.g. reactivity coefficients) and only small penalties in reactor flux.

Core conversion studies can be summarized in three main categories. These are:

1- Neutronic studies

Neutronic studies includes density search calculations, comparative study of high enriched and low enriched fuels, gradual transition to low enriched fuel core (mixed core search).

2- Safety analysis

The hypothetical loss of coolant accident (LOCA) has been the design basis accident (DBA) for TR-2 research reactor. The double ended pipe rupture or loss of piping integrity (LOPI) would remove the coolant (water) and leave only convection cooling by air, structural thermal conduction, and thermal radiation as the modes for cooling for the reactor. Therefore a great effort has been taken to determine the core conditions, safe operating power which would not yield a fuel meltdown following a LOCA. Besides LOCA the second main accident which would cause partial fuel element meltdown, is the
flow blockage in cooling channels. This is important, because of the narrow and closed cooling channels in plate type fuel elements. A flow blockage due to bowing of fuel plates or accidental dropping of foreign material into the reactor core can cause propagation of damage to the neighbouring channels and fuel plates. To provide a consistent based on fundamentals, methodical analytical approach to the problem of determining the peak temperature following LOCA or coolant channel blockage, core conversion program began an effort to develop or use existing codes to replace the frequently used single node models as a prediction tool.

The other essential part of core conversion safety analysis is the transient studies. From safety point of view three broad categories of transients are selected. These are:

- uncontrolled slow reactivity insertions that may occur during reactor startup
- loss of flow transients
- rapid reactivity insertions that may occur due to failure or malfunction of a core component or misoperation of the reactor.

For all above cases a two channel model is utilized. One channel represented the hottest plate and flow channel in the core and the second represented the average plate and flow channel.

Finally the other safety related activities consist of the thermohydraulics calculations under normal operating conditions, determination of safety margins like onset of nucleate boiling (ONB), onset of flow instability (OFI), pump coastdown, atmospheric dispersion of radioactive materials.
3- Preparation of final safety analysis report

According to the governmental rules and licensing procedures, a final safety analysis report will be prepared. Also there are some individual activities, which can be summarized as headlines as follows.

- Fuel rod behaviour under normal and abnormal operating conditions
- Flow blockage in open channels
- LOCA analysis in selected PWR type reactors by using system code RELAP
The progress in the development of any industry, including nuclear power, depends substantially on its safety for man and his environment. The most serious potential danger of nuclear power development is radioactive contamination of the biosphere. The man is one of the elements of the biosphere, which is the most sensitive to such contamination. Radioecological data demonstrate that the protection of man against radiation effects ensures usually a corresponding protection of the whole community of flora and fauna. In the context of USSR nuclear power the safety usually means the quality of nuclear power plants which as a result of the following:

- inherent design features;
- special engineering means, and
- organizational and technical measures

excludes the exceeding of the established internal and external radiation dose limits for the plant personnel and the population at a nuclear power plant, on its site and beyond the site at a distance prescribed by the regulations, as well as of norms and standards for radioactive substances concentrations in the environment both during normal operation and in the case of any accident.

From the very beginning of the construction of nuclear power plants and other nuclear facilities special attention was paid to the problem of safety. To solve the problem it was generally necessary to implement a whole set of measures, which may be categorized as follows:

- establishment of goals and corresponding criteria (norms) of safety;
- development of engineering safety means ensuring implemen-
tation of the adopted safety criteria;
- development of a wide range of administrative and organizational requirements and measures ensuring implementation of the adopted safety criteria and functioning of engineering safety means.

To develop measures for the control of safety it is necessary to create effective accident prevention means, to be able to describe accident sequences and to assess its consequences, and to find optimum means of protection and mitigation (safety means).

It is necessary to model the whole range of physical processes; the modelling includes:

1) development of computer codes;
2) checking their credibility at representative experimental devices;
3) probabilistic safety analysis and risk assessment.

These principles form the basis of the USSR research program to ensure nuclear power plant safety, that covers priority areas where further research and technical data are needed.

1. A range of hydrodynamic and heat exchange studies at reactor facilities in the case of accidents.

A comparative analysis of Soviet and foreign computer programs for safety assessment showed that the levels of computer codes development are rather close as far as their coverage of safety problems is concerned. However the main Soviet codes use a homogeneous equilibrium model of two-phase flow (some of them take into account slip ratio and partial inequilibrium), and modern foreign codes of "improved assessment" are based on complete two-fluid inequilibrium models. Modern safety approach requires codes of "improved assessment".
Studies of thermophysical and hydrodynamic processes using integral experimental rigs, which are similar structurally, hydrodynamically and thermophysically to real circulation loops of nuclear power plants with different reactor types, are a necessary element of the modelling.

The USSR carries out such studies at integral safety rigs that model circulation circuits of VVER and RBMK reactors. Now new large-scale universal safety rigs of large capacity are being created, that correspond to modelling requirements of thermohydraulics in nuclear power plants circuits with VVER and RBMK reactors, and that are oriented both at existing and future reactor facilities, and that are equipped with modern systems of instrumentation and control, and of experimental data collection and processing.

The rigs will be used for the following studies:
- thermohydraulic processes during transients and accidental conditions, including accidents with a "small" leak from a circulation loop;
- effectiveness of emergency core cooling systems;
- departure from nucleate boiling (DNB) during transients that might occur at a nuclear power plant in case of a failure of one or more main circulation pumps or an ejection of control rods from the reactor core, etc.;
- heat exchange in fuel channels models with failed fuel elements in stationary conditions with DNB and post DNB;
- conditions of post-accident heat exchange;
- diagnostics of processes that occur during accident conditions and transients;
- set of test experiments to confirm and improve computer programs.
2. Research in the area of fuel elements and assemblies behaviour in accident conditions with the following main purposes:

- tests of fuel elements and assemblies; behaviour in rig and reactor conditions to find the main factors influencing the heat transfer in accident conditions;
- detailed studies of physical and mechanical properties of fuel element components.

To achieve these aims the following studies are suggested:

1) reactor tests of VVER fuel assemblies and individual fuel elements in the conditions imitating accidents using research reactor loops to study fuel elements behaviour during loss of coolant accidents ("large", "medium" and "small" LOCAs) and during power surge accidents;

2) rig tests of fuel bundle and individual fuel element imitators of VVER and RBMK types during LOCA to study processes of deformation, heating-up, oxidation, and influence of specific parameters;

3) studies of properties of fuel cladding structural materials and fuel matrix.

3. Research of fuel behaviour and radioactive fission products release during severe accidents with fuel melting and core damage.

One of the main problems related to severe nuclear power plant accidents is the behaviour of fuel elements and radioactivity release in case of fuel element failures including fuel melting. The quantity of radioactive products released from a fuel element is a central factor for the assessment of radiation consequences of such accidents.

Mathematical modelling of the process is difficult due to complex physical phenomena occurring in such case. Such phenomena
include heat transfer conditions, fuel-cladding and cladding-coolant interactions, kinetics of cladding oxidation and rupture of materials structure changes, interaction of molten fuel with structural materials, shielding materials and the coolant. Direct experimental measurements of fission product releases in case of fuel melting are practically the only source to get credible information.

In small-scale experiments using samples it is practically impossible to reproduce some processes, such as the reaction between steam and zirconium or interaction of molten fuel with structural materials, shielding materials and the coolant, that are possible during a real accident. These processes determine the total quantity of generated non-condensable gases that influences the pressure in the containment or in some other mitigation system and the probability of their failure.

Credible information based on studies of melting processes using a special experimental facility for large-scale experiments will significantly influence the technical decisions and regulations in the area of safety and, finally, the nuclear power plant economics and nuclear power development as a whole.

These include:
1. Requirements to emergency cooling systems effectiveness.
2. Requirements to mitigation systems effectiveness.
3. Siting of different-purpose nuclear power plants around towns and generally on the territory of the country.
4. Emergency planning to protect the public.

Moreover it is planned to carry out experiments using loops of research reactors to study fission product releases in simulated accident conditions, to construct a special rig for experimental research of physical and chemical forms and compounds of
fission products as a result of their interaction with reactor materials and for testing the effectiveness of emergency localization and mitigation systems.

4. Problem of hydrogen at nuclear power plants.

The presence of hydrogen in water cooled nuclear reactors is a reality that should be taken into account.

In design-basis normal and accident conditions without reactor core superheating (melting-down), that is necessary for chemical interaction between steam and reactor structural materials, the presence of hydrogen (or explosive mixtures) is not usually dangerous and does not cause sequences leading to a severe radiation accident.

Hydrogen-related safety in design-basis events is ensured by design. An example of this approach is the design of RU AST-500 reactor facility for central heating.

In a hypothetical accident the generation of a large quantity of hydrogen may lead to very severe consequences that will determine the nature of the accident sequences and the level of the damage. In such cases even under the assumption that it is impossible to mitigate hydrogen effects, the post-accident measures should take into account the possible consequences of hydrogen behaviour. For this purpose in order to assess the accident consequences it is necessary to study the whole range of the hydrogen problems including hydrogen generation and ignition (explosion).

The following questions are planned to be studied within the framework of the hydrogen problem research program:

- hydrogen generation in radiolytic and chemical processes,
- hydrogen distribution in the reactor circuit and plant rooms,
- characteristics of hydrogen burning,
- methods of hydrogen detection,
- methods of mitigation of hydrogen influence on accident development,
- viability of safety systems and equipment,
- legislation and regulations development.

5. Reliability analysis of nuclear power plant systems and components, research on probabilistic safety analysis and risk assessment.

At present the improvement of nuclear power plant safety is connected with application of probabilistic safety analysis. The analysis ensures a balanced approach to safety decision making process during nuclear power plant design, operation and regulation. Its application permits: to analyze nuclear power plant safety; to identify weak elements in nuclear power plant systems, the reliability of which depends on a large number of components; to justify technical and organizational safety measures; to determine reliability requirements for systems and equipment; to develop optimum procedures of their maintenance; to train operators; to determine preventive measures for hypothetical accidents; to set priorities and effectively use resources to ensure safety; to determine inspection activities; to justify regulations; to present a general quantitative assessment of nuclear power plant safety.

Disadvantages of probabilistic approach utilization are determined mainly by: insufficient knowledge of the plant personnel behaviour reliability in normal operation and accident conditions; existing uncertainty in analysis results and difficulties in their processing; incomplete modelling of accident processes; incomplete data base on reliability of nuclear power
The proposed research is connected with the following studies:

a) determination of primary failure probabilities (probabilities of accident initiators);

b) development of possible accident sequences taking into account the operation of service safety systems;

c) determination of safety systems failure probabilities for given accident sequences;

d) analysis of physical, hydrodynamic, mechanical, chemical and other processes during accidents;

e) determination of composition and quantities of radionuclides escaping the nuclear power plant;

f) determination of radionuclide distribution beyond the nuclear power plant;

g) assessment of consequences for the population taking into account meteorological conditions and population distribution;

h) development of probabilistic indexes of nuclear power plant safety and choice of criteria values.

The following studies are planned as well:

- development of scientific basis for safe control of technological processes at nuclear power plants;

- development of technical means and equipment diagnostic systems;

- nuclear safety studies for fuel transportation;

- studies of processes and hydrodynamic effects in localization systems during accidents with a rupture of primary circuit;

- development of norms and standards for radiation, medical and ecological effects of nuclear power plants;
- strength and rupture dynamics studies of reactor facilities structures and equipment.

Considerable attention will be paid to studies of hypothetical accidents with a possible destruction of protective barriers and fuel melting in order to assess their probabilities and radiation consequences. The studies should contribute to the development of proposals on the design of stand-by technical means for nuclear power plants aimed at mitigation of severe accident consequences and on maximum possible release of radioactive substances to be taken into account in nuclear power plant siting and development of population protection measures.

The implementation of the above program should permit:

- to carry out a comprehensive safety analysis of existing and future nuclear power facilities and to justify norms and standards in this area at a qualitatively new level;

- to create an experimental basis with modern computers and instrumentation to solve problems of improved reactor facilities with increased safety;

- to choose scientifically justified criteria for nuclear power plant safety and to formulate a national concept on the issue.
The CEGB is building a large PWR at Sizewell which is intended to be the first of a series of similar plants. The reactor will be of Westinghouse design. The Central Electricity Generating Board (CEGB), the utility which supplies electricity to the whole of England and Wales, has decided to opt for a PWR in recognition of its strong world-wide commercial position.

Before giving approval, the government ordered a public inquiry under the Department of Energy. This covered aspects of need, economics, safety and environmental impact. The inquiry was presided over by an Inspector, a lawyer, aided by four technical advisers. It started at the beginning of 1983 and the hearings ended early in 1985. After further delays a weighty report from the Inspector was placed on the desk of the Minister of State for Energy at the end of 1986. The report was published in January 1987 and recommends that approval be given for construction of the reactor. On 12 March 1987, the Minister announced that he was giving this approval.

With this background, the safety research work has been concerned with:

(a) the work to establish the case for the pre-construction safety report (PCSR);

(b) the more severe accidents within the design basis which might have an effect on the detail of the reactor design;

(c) planning for the provision of the necessary support for licensing and operation.

LICENSING IN THE UK

Licenses to operate are granted by the Nuclear Installations Inspectorate (NI). In contrast to the USNRC, this is a very small body, being only about 120 strong in total. Responsibility for safety rests very firmly with the operating utility. The NI lays down general guidelines for acceptability in the form of safety principles, but it is up to the utility to make its detailed safety arguments using such methods as it deems to be appropriate. The NI then considers the safety document on its merits, asks detailed questions, may request changes and, if it is satisfied with the case offered, grants a licence. Since the utility is totally responsible for safety, it requires a strong technical capability to be able to satisfy itself on the safety of the plant. The PCSR for Sizewell B has been largely based on US practice. For LOCA, for instance, the Appendix K approach is used. It is noteworthy however that the Inquiry Report stresses the desirability of moving towards a case based on best estimate calculations.

RESPONSIBILITIES

In this section we list briefly the roles of the various organisations involved in design, construction, licensing and operation of reactors in the UK.

NNC - The National Nuclear Corporation.

This is the company which designs and builds reactors in the UK. For the PWR, NNC have a Westinghouse licence so that the Sizewell B design is a 1200 MW(e) Westinghouse design with some local modifications. NNC are also contracted to write the safety document and to undertake the necessary supporting analysis.
CEGB - Central Electricity Generating Board.

Owns and operates all power stations and the distribution network in England and Wales. It is the customer for Sizewell B. It is a very large utility with strong internal technical support. It takes steps to see that in contracting for new plant, it is an informed buyer on all technical issues.

BNFL - British Nuclear Fuels PLC.

This is the UK company responsible for fuel design and supply. For PWR fuel, they have a licence from Westinghouse Nuclear Fuels Division and carry out neutronics and core thermal-hydraulics design calculations. They are contracted to supply the first fuel load for the Sizewell Project.

UKAEA - The United Kingdom Atomic Energy Authority.

This carries out a wide ranging research and development function. In the PWR context it serves both to support the CEGB and to give independent advice to government agencies and the NII.

NII - The Nuclear Installations Inspectorate.

This is the licensing authority for civil nuclear installations.

R&D PROGRAMME

In the UK an integrated R&D programme has been set up to support the construction, safety, licensing and operation of the PWR. The work is spread over the various laboratories of the CEGB, the UKAEA, NNC, BNFL and other contractors. The current areas of work include:

- Reactor Physics
- Fuel activity migration
- Heat transfer and hydraulics
- Materials
- Severe accidents
- Chemistry studies
- General Nuclear Safety Research, covering mainly:
  - Degraded core studies
  - Fuel coolant interactions and fuel material studies
  - Probabilistic risk assessment and reliability requirements

ACKNOWLEDGEMENT

This paper is an extract from a paper by Dr I H Gibson of Atomic Energy Establishment, Winfrith.

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UKAEA
Summary of Research Activities in Progress
(Including International Cooperation Highlights)

1. Introduction

The NRC has the responsibility to assure that nuclear power plants are designed, constructed and operated in a manner which is consistent with the protection of the health and safety of the public. The NRC operates according to the principle that safety of plant design, construction and operation is the responsibility of the licensee. Nevertheless, the NRC staff must have the ability to independently assess plant designs and safety analyses submitted by license applicants and to review operating experience. This requires foremost that a sound physical understanding be obtained through research of the important phenomena that may occur in operating power plants.

The purpose of research is to produce the information necessary for the NRC to carry out its regulatory function. The principal objective of research is to understand the behavior of light water reactors (LWRs) during all normal and upset conditions, so that regulatory actions affecting design, operation, and maintenance can be made on a firm technical basis. The following summarizes the main aspects of the NRC research program associated with power reactors.

2. Primary Circuit Integrity

As part of the assurance of reactor safety, it is vital to have confidence that leaks or breaks will not develop in the reactor coolant system including the reactor pressure vessel and associated piping. Small defects initially present in thick steel components can grow under the stresses arising from repeated pressure and temperature changes, and from embrittlement due to high energy neutron radiation damage.

The Plate Inspection Steering Committee (PISC-I) program was originally established to assess the capabilities of the 1974 ASME Code Section XI ultrasonic procedure to detect, size and locate flaws in heavy section steel. PISC-I was carried out from 1976 to 1980. This program was followed by PISC-II to perform a more extensive evaluation of the best performance obtainable by modern ultrasonic techniques under optimal conditions. PISC-II was formulated to examine in more detail which techniques could prove the desired level of capability for detecting and sizing defects. PISC-II, which lasted from 1980 to 1986, included participants from the European Economic Community, Japan, and seven inspection teams from the United States.
The current program, PISC-III, includes the capability and reliability to detect and characterize cracks in pressure vessels, stainless steel piping, and nozzles using real components with real flaws under realistic field conditions. Included in the program are structured human reliability studies to quantify the parameters that have the greatest effect on reliability and to make recommendations for improvements in training and procedures.

Piping degradation over the past decade or more has had a significant impact on the reliability of plant operation and has resulted in personnel exposures for inspections and repairs and high costs for remedial actions. The assumption of instantaneous double ended guillotine break has lead to expensive requirements for pipe whip restraints, component supports and jet impingement shields. These provisions make in-service inspection for cracks difficult due to inaccessibility, with significant radiation exposure to inspectors. RES, therefore, organized the International Piping Integrity Research Group (IPIRG) as a cooperative program on piping experiments. The purpose is to develop the technical basis for the rule change to General Design Criterion 4 to replace the double ended break criterion. IPIRG consists of four major tasks: (1) leak before break assessment of piping subjected to dynamic seismic forces; (2) development of experimental data base on material property and pipe fracture; (3) fracture of piping containing high pressure fluids; and (4) resolution of unresolved issues from Phase I of the NRC degraded piping program. IPIRG complements the results of Phase II of the NRC degraded piping program.

Research on steam generators is focused on the Steam Generator Group Project at the Hanford Engineering Development Laboratory (HEDL). The project utilizes a former Surry steam generator. The project participants are France, Italy, Japan, Electric Power Research Institute (EPRI), and the NRC. The purpose is to develop validated models for predicting margins to failure under burst and collapse pressures for degraded steam generator tubes. This information is used to determine appropriate criteria for when tubes should be plugged. The steam generator program also investigates the reliability of nondestructive examination (NDE) for characterize tubing flaws, evaluation of advanced NDE equipment, and optimizing NDE procedures. In this way, a correlation can be developed between the flaw signal and tube integrity.

Neutron dosimetry research is conducted to establish reliable methods for measuring the fluence to reactor pressure vessels and to study possibilities for lowering the fluence. International cooperative efforts in neutron dosimetry have been underway since the late 1970s. Through these efforts, the NRC has gained information and expertise in measurements and calculation of neutron spectra and fluxes in experimental rigs both in the NRC programs at HEDL, Oak Ridge National Laboratory (ORNL), and the National Bureau of Standards (NBS), and from abroad, particularly Belgium the United Kingdom. Many utilities are utilizing dosimetry in the vessel cavity as an alternative to in-vessel dosimetry. The reliability of these measurements as indicative of vessel fluence exposure must be evaluated. In this regard, the NRC has established a cooperative agreement with the United Kingdom. The NESTOR Shielding and Dosimetry Improvement Program (NESDIP) is being carried out by the United Kingdom Atomic Energy Authority. NEDSIP is concerned with the study of benchmarks of the radial shield, cavity and nozzle regions. The purpose is to produce a benchmark for calculation and measurement accuracy of reactor vessel fluence and for the treatment of cavity streaming. The NRC
contributes assistance with measurements and analysis and plant data from H.B. Robinson and Davis Besse. A program in the Belgian VENUS facility is producing data to benchmark predictions of flux for cores containing burned or dummy fuel elements at the core periphery as a means of reducing the fast fluence to the reactor vessel (the scheme being employed at H.B. Robinson, for example, is calculated to reduce the vessel fluence by an order of magnitude). The NRC contributes assistance with measurements and analysis. The experiments are performed by the United Kingdom.

A cooperative program has been established among the NRC, F.R. Germany, and the United Kingdom as the principal partners, but with other countries participating on an ad hoc basis. The program is devoted to the study of pressure vessel material from the German Gundremmingen KBK-A reactor. Properties are to be determined for the as-irradiated and annealed conditions. Unirradiated properties will be determined from material taken from a region where no neutron irradiation occurred. Complementary testing is being performed on archive material specimens. Test reactor irradiation will also be conducted in the United Kingdom and the United States to evaluate whether a dose rate effect exists. The United Kingdom is in addition conducting a number of studies of the mechanisms of embrittlement in both the actual vessel material and in the archive material.

Research on plant aging is intended to resolve issues related to the aging and service wear of equipment and systems and their possible impact on degrading the level of plant safety. Aging includes both wear of moving parts in components, corrosion processes, and degradation of mechanical properties such as embrittlement of plastic insulation. To evaluate phenomena associated with aging on nuclear power plant components, RES needs information on in-service degradation processes in components and systems and naturally aged components for performing post service examinations and testing. In this regard, components are being obtained through DOE from the decommissioned Shippensport power station. Information from domestic sources is being supplemented by cooperative research programs with foreign organizations.

3. Containment Integrity

At Sandia National Laboratories (SNL), RES sponsors an analytical and experimental program to determine the mechanical strength and failure modes of containment structures when subjected to severe accident conditions of pressure, temperature and seismic loads. The program concerns: (1) failure of the containment shell or liner due to overpressurization; (2) leakage at large penetrations as a result of deformation of the structure or degradation of seals and gaskets; (3) leakage at electrical penetrations due to degradation of seals subjected to high temperature and pressure; and (4) leakage through valves.

Thus far, pressure tests have been run on steel containment models. Four 1/32 size steel models (1.1 m diameter, 1.65 m height) of three configurations were tested at SNL. One of the four contained penetrations simulating personnel and equipment hatches. One test was performed on a 1/8 scale steel model with simulated personnel and equipment hatches and a constrained pipe penetration. An reinforced concrete containment model of 1/6 scale (7 m diameter, 11 m height) is currently being prepared for pressure testing. In a related effort, EPRI is conducting large scale separate effects tests on containment
penetrations and on basement-to-wall junctions. RES has not planned any tests on prestressed concrete containment models. The United Kingdom is considering the construction and testing of a model prestressed Sizewell containment. Program participants perform pretest predictions. The tests and analysis determine how containments respond to severe accident conditions and the availability of analytical methods to calculate the behavior.

4. Seismic

Seismic safety research is carried out to determine the probability of occurrence of an earthquake of a given magnitude and effects of earthquakes induced forces on nuclear power plant equipment and structures. This includes the ability to calculate the stresses produced on equipment and structures for a given frequency and amplitude of vibratory ground motion. Experiments on piping systems have shown that realistic damping values are considerably higher than the values used in existing design. Fracture mechanics and reliability models show that flexible piping is more reliable than the currently required rigid piping, particularly when the unreliability of snubbers is considered. Removal of unneeded pipe supports and snubbers permits considerable cost savings while increasing plant safety.

International cooperation is carried out in the conduct of experiments on the response of structures and piping systems to earthquake loads and the validation of computer codes used to predict the response of plant structures and systems. In Japan, a massive testing effort was created in 1974 under the sponsorship of the Japanese Ministry for International Trade and Industry. The Nuclear Power Engineering Test Center was established and the largest shake table facility in the world was constructed at the Tadotsu Engineering Laboratory. The table is 15 m x 15 m with a capacity of 1000 tons. By comparison, the largest shake table in the United States is operated by the Richmond Field Station of the University of California with a 6 m x 6 m size and a 60 ton capacity. A series of eight experiments are being carried out from 1982 to 1988 to study the response of containment vessels, primary loops, reactor vessels, and reactor internals for both PWRs and BWRs. All specimens will be excited with time histories representative of the Japanese design basis earthquake, designated S1 and S2. The designations are similar to the NRC designations of Operating Basis Earthquake and Safe Shutdown Earthquake. The responses to the induced motion will be monitored. The total cost of the program exceeds $500 million, including about $200 million for the facility.

RES is cooperating with F.R. Germany on programs in the Heissdampfreaktor (HDR) facility. HDR is a decommissioned nuclear power plant located in Kahl, F.R. Germany that has been used for a number of safety studies. The current Phase II experimental program is being performed from 1984 to 1988. The HDR SHAG test series was completed in July, 1986 and is currently being evaluated. The SHAG tests measured the responses of piping systems when the internal containment structure was excited with a large mechanical shaker. Starting in late FY 1987, a series of experiments (termed SHAM) will be performed in which one of the piping systems will be excited at frequencies and amplitudes typical of severe seismic events, with acceleration forces exceeding SSE levels. A system of shakers will be used to excite the piping in the X, Y, and Z directions simultaneously. The SHAM experiments will provide data for validation of computer codes used to predict the behavior of piping into the inelastic range. The tests will also evaluate the effects of various piping
flexibilities and flow forces on the operability of a typical gate valve. This will be used to identify loading combinations important to the qualification of safety related equipment.

Joint cooperation is being carried out with France to exchange data and discuss the analysis of data from an experiment to measure the propagation of seismic energy in the immediate vicinity of a recorder site within 100 to 200 m of a power plant foundation. The data are being collected at two sites in California.

RES is participating in a joint program with Taiwan, where a 1/6 scale concrete containment model is being constructed in a seismically active area in Taiwan. In fact, an earthquake of Richter magnitude 7 recently occurred within 60 km of the site. The model is instrumented and recordings of earthquake will be made over a 5 year period from 1986 to 1990. RES will perform low level vibratory tests of the model to provide baseline data on modal parameters. Analytical models of soil-structure interaction will be compared with the recorded observations.

In eastern and central United States the understanding of the potential for earthquakes is poor as compared to the west, where it has been extensively studied. Thus, RES has been collecting information from four eastern regional seismic networks operated in cooperation with universities and State and Federal agencies. These four networks include 240 stations. Cooperation is underway with the Canadian Geological Survey on the regional seismographic network of the northeastern United States.

5. Thermal-Hydraulic Transients

The principal products of thermal-hydraulic research are analytical tools (computer codes) to understand and predict the plant response to disturbances from normal operating conditions. The codes model the plant behavior by describing the processes of heat transfer and fluid flow that occur in a plant. The simulation of the expected behavior of the fluid systems in LWRs resulting from transients is normally performed with computer codes since it is too costly and unsafe to test the response to severe events using actual power plants.

Whenever possible, physically-based modeling is used in the development of the codes. Empirical modeling developed from thermal-hydraulic experiments is used to characterize important phenomena. Such experiments are conducted in integral and separate effect facilities. The NRC has had the ability to perform integral systems tests in facilities representative of each major LWR type, i.e., Westinghouse and Combustion Engineering PWRs (Loss of Fluid Test (LOFT), SEMISCALE), Babcock and Wilcox PWRs (Multi-loop Integral System Test (MIST)), and General Electric BWRs (Full Integral System Test (FIST)). These facilities were all designed from the same basic principles and included features necessary to simulate plant transient responses. The final test in LOFT was conducted in July, 1985. SEMISCALE was shutdown in March, 1986 and is being kept test-ready temporarily. MIST is scheduled to shutdown following test completion in FY 1987 or FY 1988, depending on agreement on a follow-on program. FIST has been shutdown for two years.
RES has cooperative arrangements concerning many foreign thermal hydraulic facilities including Rig of Safety Assessment-IV (ROSA-IV), Cylindrical Core Test Facility (CCTF), and Slab Core Test Facility in Japan and Upper Plenum Test Facility (UPTF) in F.R. Germany. ROSA-IV is a four year cooperative program that began in 1984 between the Japan Atomic Energy Research Institute (JAERI) and the NRC involving safety research on small break loss-of-coolant accidents (LOCAs) and transients in PWRs. The experimental program began in May, 1985 and will continue until 1988. ROSA-IV consists of the Large Scale Test Facility (LSTF) and the Two-Phase Test Facility (TPTF), which are being used to obtain integral and separate effects data, respectively. ROSA-IV is the largest scale and most important single facility in the world for the study of small break LOCAs and transients in PWRs. The facility is scaled 1:50 in terms of power and volume compared to a full size plant, and 1:1 in terms of vertical scale. By contrast, SEMISCALE was scaled 1:1700 in terms of power and volume, 1/35th the size of ROSA-IV. The size of ROSA-IV is similar to LOFT but whereas the design of LOFT was oriented toward the study of large break LOCAs, ROSA-IV is aimed at the study of less severe, but more probable transients.

The 2D/3D program is a three party cooperative program among the NRC, JAERI, and the German Federal Ministry for Research and Technology (BMFT) for the study of large break LOCAs in PWRs. The program was initiated in 1978 and will continue until 1988. The program includes three large scale experimental facilities: CCTF, SCTF, and UPTF. The program is producing a large body of experimental data on heat transfer and fluid flow for the study of two and three dimensional flow behavior during the refill and reflood phases of a large break LOCA. The data are used for development and assessment of TRAC-PWR in its ability to model core cooling following pipe breaks in the reactor coolant system. The 2D/3D test facilities are the largest in the world used for studying the refill and reflood processes of a PWR LOCA. As such, they are used to determine whether the phenomena observed in small scale facilities also occur in larger facilities and to supplement and complement LOFT and SEMISCALE results. With the scalability of the test data confirmed, the extrapolation to a full size reactor using computer codes is more reliable.

It is generally recognized that the safety assessment of LWRs requires the development of advanced thermal-hydraulic computer codes for use in safety related studies. This common interest led RES to organize an international cooperative program to undertake code assessment and application that is intended to:

- Provide a statement on code performance;
- Share user experience on code validation;
- Share experience on code errors and inadequacies; and
- Establish and improve user guidelines for applying the code.

The sharing of effort by the international nuclear community allows these goals to be met through an optimum use on limited resources. The cooperative effort is planned to continue over a period of approximately five years, from 1986 to 1990. Participating organizations share information relating to
analysis methods and code uncertainties. The NRC furnishes its advanced, best-estimate thermal-hydraulic codes, test data, and code assessment results in exchange for code assessment studies and the experimental data necessary for code verification.

The International Code Assessment and Applications Program (ICAP) participants perform assessments using data from their experimental facilities and power plants and provide the results to the NRC. The NRC uses this information and the results of domestic assessment and application of the code to arrive at overall conclusion on the accuracy and reliability of the codes for best-estimate predictions of plant behavior. The size of this uncertainty is used in deciding when code improvement should be considered to be completed. The assessment effort includes a minimum validation matrix for thermal hydraulic codes formulated through the OECD Nuclear Energy Agency and the Committee on the Safety of Nuclear Installations.

Cooperation is carried out with Belgium, Finland and F.R. Germany on thermal mixing following ECC injection. The Imatran Voima Oy utility of Finland performed a series of tests in a 1:25 scale low pressure experimental facility consisting of a semiannular downcomer and three cold legs. The results were analyzed with RES thermal mixing codes.

6. Severe Accidents

Operating reactors are licensed in accordance with 10 C.F.R 100.11 which requires that the exposure at the site boundary not exceed 25 rem in the event of an accident. The assumed fission product source term release to the containment for the purpose of containment leakage analysis is taken from TID 14844. Containment loadings in terms of pressure and temperature, however, are taken from Design Basis Accidents such as large break LOCA.

Following the Three Mile Island-2 accident, the research effort devoted to severe accident phenomenology and fission product behavior was greatly increased. Major research experiments did not actually begin until late FY 1982 since lead times for such complex efforts are considerable. The basic requirement was the development of a calculational methodology such as a set of computer models which allow for the computation of fission product release, transport, deposition, and location throughout the reactor coolant system and containment for the range of postulated severe accident sequences. In addition, the pressure and temperature of the reactor coolant system and containment must be evaluated as a function of time. An analytical treatment of physical phenomena requires the input of physical modeling for parameters such as diffusion coefficients, heats of vaporization, oxidation rate constants, heats of chemical reactions, conductivity, mechanical properties, and thermal-hydraulic behavior. All such models have uncertainties associated with them due to data scatter, extrapolation of values outside the range of known values, experimental error, basic experimental limitations, and scaling. The severe accident research attempts to arrive at best estimate modeling based on physical understanding of the important phenomena and to evaluate the uncertainties resulting from inexact knowledge of these phenomena. The principal codes that have been developed include:

MELCOR: Simplified, relatively fast running code for analysis of
Severe accident research is carried out through the Joint International Severe Fuel Damage and Source Term Research Program, which was organized by RES. In addition to the NRC, it includes EPRI and 11 foreign countries. Participants furnish funding and in-kind research contributions. In addition, cooperation is maintained with the Industry Degraded Core Rulemaking Group (IDCOR).

In-pile severe fuel damage experiments have been carried out through the OECD LOFT Project and Power Burst Facility (PBF) at the Idaho National Engineering Laboratory (INEL), the Annular Core Research Reactor at SNL, and at the National Reactor Universal (NRU) in Canada. A total of 5 severe fuel damage experiments are being performed in NRU, of which four have run to date.

The THI-2 results obtained through investigations under the General Public Utilities/Department of Energy/NRC/EPRI arrangement are also utilized. These have been supplemented by out-of-pile laboratory studies of the UO₂-zircaloy-oxygen system and the CORA melt progression experimental program in F.R. Germany. Out-of-pile fission product release experiments are carried out at ORNL and Battelle Columbus Laboratory.

The generation and behavior of aerosols and fission products in the reactor coolant system has been studied in several programs in addition to the information obtained through LOFT, PBF, ACRR, and NRU. Studies were conducted in
the United Kingdom and F.R. Germany on the generation of aerosols from Ag-In-Cd control rods, a potentially important mechanism of fission product transport. The results showed that Cd was the only significant source. The MARVIKEN program, carried out at Studsvik, Sweden, studied aerosol transport and deposition in a full scale reactor coolant system. The interaction of fission products with aerosols and aerosol behavior is also being studied at SNL and ORNL.

Core-concrete interaction is studied in the SNL Large Melt Facility and the BETA facility in F.R. Germany. Debris bed coolability is studied at Brookhaven National Laboratory (BNL). SNL is performing large scale steam explosion experiments.

RES and EPRI cooperated in large scale hydrogen mixing and burning studies at the Nevada test site. A total of 40 experiments were performed in a 52 ft diameter vessel. The Hydrogen Control Owners Group has performed 1/20 scale experiment and is currently performing 1 scale experiments in a facility representative of a BWR Mark III containment. Research has been performed in the U.S. at the SNL Flame Acceleration Measurement Experiments (FLAME) facility. Additional research has been performed in Canada, F.R. Germany, and Norway on flame acceleration and the potential for transition to detonation. Significant progress has been made on understanding the potential for detonation.

The United States does not have any fuel irradiation facilities to study fuel behavior under commercial reactor conditions. This has necessitated the establishment of cooperative arrangements on the part of the NRC involving the OECD Halden Project in Norway, the BR-3 reactor in Belgium, the HFR Petten reactor in the Netherlands, the Studsvik reactor in Sweden, and the RISO reactor in Denmark. Most of this work was carried out during the 1970s. However, it has been necessary to purchase irradiated fuel rods from the BR-3 reactor under the Severe Accident Research Program program for use in PBF, ACRR and TREAT.

6. Probabilistic Risk Assessment

Probabilistic risk assessment is used for regulatory applications including:

- Implementation of the NRC Severe Accident Policy Statement;
- Implementation of NRC safety goal policy;
- Use with the NRC backfit rule;
- Evaluation or regulatory requirements such as emergency preparedness, plant siting, and equipment qualification; and
- Establishment of risk-oriented priorities for allocating agency resources.

A major report (NUREG-1150) has been issued in draft form in February, 1987 that provides a comprehensive assessment of risk for five plants. RES is currently seeking comments on its content and how best to present the informa-
The document describes:

- The major factors related to internally initiated events that contribute to severe core damage;
- The frequencies and associated uncertainty ranges of severe fuel damage events;
- The severe accident phenomena that could lead to containment failure and the probability and uncertainty of containment failure as a function of time;
- The consequences of severe accidents, including the effect of evacuation or sheltering;
- Comparison of risks to the NRC safety goal; and
- The costs and benefits of plant specific measures to reduce risk.

The report represents another step forward in the characterization of risk from nuclear power plants. Since the 1950s, attempts have been made to estimate the effects of postulated accidents. The first substantive attempt to develop an estimate of the potential releases arising from severe accidents is found in the AEC's WASH-1400 report published in 1957. At the time, there was a general absence of information or methods to make systematic estimates of system failure probabilities and the consequences of severe accidents. The next milestone was TID-14844 in 1962 which considered a maximum credible accident of a contained core melt and arrived at an estimated source term to the containment atmosphere of 100% of the noble gases, 50% of the iodine, and 1% of other fission products. These release quantities are referenced in 10 CFR Part 100 for determining the required containment leak-tightness.

The first major effort at a probabilistic safety assessment was documented as the Reactor Safety Study (WASH-1400), published in 1975. This study arrived at release categories associated with various levels of plant damage. The analysts in the Reactor Safety Study were frequently faced with paucities of data for fission product release, transport, and deposition processes. Consequently, simplified models were used with conservative assumptions to cover gaps in knowledge.

Following the Three Mile Island-2 accident, the need was recognized to examine the state of knowledge on source terms more closely in order to be able to make technically sound regulatory decisions. The technical bases for estimating source terms was performed and resulted in NUREG-0772 published in 1980. Concurrently, the Reactor Safety Study Methodology Applications Program defined a spectrum of accidents with varying release quantities (NUREG-0771 and NUREG-0773). These formed the basis for the Severe Accident Research Program that has been carried out over the past several years. Due to the significant progress made in the intervening years and to define remaining research needs, the report Reassessment of the Technical Bases for Estimating Source Terms (NUREG-0956) was issued in 1986. This information, and other related plant-specific risk studies, provided the technical basis for developing NUREG-1150.
Since the Reactor Safety Study, the structure and discipline of probabilistic methods have led to developments which for the first time permits logical appraisal of uncertainties and of deficiencies in our understanding. For example, detailed containment event tree and uncertainty analyses were performed and the probability of core damage, containment failure, and offsite consequences are presented in terms of a range of values within which the true value would likely reside rather than just a single point estimate of the mean. These ranges evolved form explicit consideration of uncertainties. NUREG-1150 is also the first study to use the Source Term Code Package and the consequence code MACCS, which contains significant improvements in the treatment of phenomena following release from containment. The technical basis for the assessing risk is now on a much sounder footing. Future research is aimed reducing the widths of the uncertainties bands and better defining the expected mean values.

7. Waste Management

The Nuclear Waste Policy Act provides for the development of geologic repositories for the permanent disposal of high level radioactive waste. The NRC is responsible for licensing these facilities. Research is intended to provide the technical basis for development of regulation for the review and licensing of high level waste repositories. The overall system performance standard issued by the Environmental Protection Agency for long term (10,000 years) releases to the environment has required the development of predictive models of repository performance. The research includes the overall waste package and the characteristics of the geologic repository. This includes:

- Waste packaging materials performance and failure mechanisms;
- Behavior and performance of site backfill materials such as clay;
- Groundwater flow;
- Characteristics and long term performance of conditioned high level waste and transuranic waste;
- Radionuclide migration modeling and experimental results;
- Methods of classification, treatment and disposal of low level radioactive waste including establishment of de minimis levels and results of waste form characterization experiments; and
- Methods and assessment of operational safety of waste disposal sites.

Research on low level waste includes radionuclide migration through soils. This includes transport pathways, rates, and mechanisms and the effects of soil types and chemical parameters on radionuclide migration for various radionuclide chemical forms. High level waste management research includes: determination of long term solubility of glass; corrosion tests in gamma fields for packaging materials; and characterization of natural barriers.
LONG-RANGE RESEARCH PLAN

FUNDING LEVELS*

FY 1987-FY 1991

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<td>$103.6</td>
<td>$109.5</td>
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* Dollars in millions.
NUCLEAR REGULATORY SAFETY RESEARCH

JOSÉ LUIS M. CORTEZ
OFFICE OF NUCLEAR REGULATORY RESEARCH
U.S. NUCLEAR REGULATORY COMMISSION

TECHNICAL COMMITTEE MEETING ON THERMAL REACTOR SAFETY RESEARCH
INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, AUSTRIA

JUNE 9-12, 1987
OFFICE OF NUCLEAR REGULATORY RESEARCH

1. PROGRAM - INTEGRITY, OPERABILITY AND AGING OF STRUCTURES AND COMPONENTS

THIS PROGRAM CATEGORY IS DIRECTED TOWARD ENSURING REACTOR AND PLANT COMPONENTS AND SYSTEMS PERFORM AS DESIGNED AND THAT INTEGRITY OF SUCH ITEMS IS MAINTAINED OVER THE LIFE OF REACTOR AND PLANT.

<table>
<thead>
<tr>
<th>ELEMENT</th>
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<tr>
<td>REACTOR PRIMARY SYSTEM INTEGRITY</td>
<td>PRESSURE VESSEL SAFETY AND PIPING INTEGRITY</td>
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<td>ELECTRICAL AND MECHANICAL COMPONENTS</td>
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<td>MAINTENANCE, REPAIR, REPLACEMENT</td>
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<td>COMPONENT RESPONSE TO EARTHQUAKES</td>
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<td>VALIDATION OF SEISMIC ANALYSIS AND SEISMIC DESIGN MARGINS METHODS</td>
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<tr>
<td>REACTOR CONTAINMENT INTEGRITY</td>
<td>STRUCTURAL TESTS</td>
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<td>VALIDATION OF PREDICTION METHODS</td>
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<td>ENGINEERING ISSUES</td>
<td>RESOLUTION OF USIs/GSIs</td>
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</table>
2. PROGRAM - NPP SYSTEMS SAFETY FUNCTION RELIABILITY

RESEARCH IN THIS PROGRAM CATEGORY IS DIRECTED TOWARDS DEVELOPING AN UNDERSTANDING OF THE BEHAVIOR OF REACTOR SYSTEMS IN SEVERE TRANSIENTS WITH THE OBJECTIVE OF ENSURING THAT SAFETY SYSTEMS WILL PERFORM THEIR INTENDED FUNCTIONS ADEQUATELY.

<table>
<thead>
<tr>
<th>ELEMENTS</th>
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<td>HUMAN FACTORS INSPECTION GUIDANCE</td>
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<td>PLANT &amp; SYSTEMS RISK AND RELIABILITY</td>
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<td>REACTOR SYSTEM ISSUES</td>
<td>DEPENDENT FAILURE ANALYSES AND DATA COLLECTION</td>
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<tr>
<td>ACCIDENT MANAGEMENT</td>
<td>INDIVIDUAL PLANT EVALUATIONS</td>
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3. PROGRAM - PUBLIC PROTECTION FROM RADIATION

This program category involves research to better understand severe accident risk and phenomenology to ensure that the reactor systems are designed to be adequate and reliable to mitigate the effects of "loss-of-core cooling".

<table>
<thead>
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<th>ELEMENT</th>
<th>ACTIVITY</th>
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<td>SOURCE TERM EXPERIMENTS</td>
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<td>CONTAINMENT LOAD EXPERIMENTS</td>
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<td>REACTOR ACCIDENT RISK</td>
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<td>METHODS &amp; ANALYSIS</td>
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<td>RISK APPLICATIONS/IMPROVEMENTS</td>
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<td>SEVERE ACCIDENT POLICY IMPLEMENTATION</td>
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4. PROGRAM - SAFE CONFINEMENT OF RADIOACTIVE WASTE

Research in this category is directed toward developing a competent technical base and calculational methods to allow NRC to review, evaluate and license DOE facilities for high level waste disposal and to assist state and local governments in responding to the Low Level Waste Policy Act of 1985.

<table>
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<th>ELEMENT</th>
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<td>● HYDROLOGY &amp; GEOCHEMISTRY</td>
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<tr>
<td></td>
<td>● COMPLIANCE, ASSESSMENT AND MODELING (HOW WE ASSESS APPLICANTS' PROGRAM APPROACH)</td>
</tr>
<tr>
<td></td>
<td>● TECHNICAL SUPPORT TO REGULATION</td>
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</table>

| LOW LEVEL WASTE MANAGEMENT     | SAME AS HLW                                                               |
OFFICE OF NUCLEAR REGULATORY RESEARCH

5. PROGRAM - GENERAL REGULATORY SUPPORT AND REGULATION DEVELOPMENT


THIS CATEGORY ALSO SUPPORTS EFFORTS TO ENSURE THAT PROPOSED COMMISSION REGULATIONS ARE COST EFFECTIVE AND THAT THEIR DEVELOPMENT IS CARRIED OUT IN AN EFFICIENT AND TIMELY MANNER.

<table>
<thead>
<tr>
<th>ELEMENT</th>
<th>ACTIVITY</th>
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| - NON-REACTOR RESEARCH AND STANDARDS DEVELOPMENT | - FUEL CYCLE  
- TRANSPORTATION/SAFEGUARDS 
- OCCUPATIONAL RADIATION PROTECTION AND HEALTH EFFECTS  
- MATERIALS SAFETY |
| - STANDARD ADVANCED REACTORS | - REVIEW DOE ADVANCED REACTOR CONCEPTS  
- ESTABLISH LICENSING CRITERIA 
- PLANT STANDARDIZATION 
- QUALITY ASSURANCE STANDARDS |
| - REGULATORY IMPROVEMENT | - DEVELOP OR MODIFY REGULATION  
- INDEPENDENT REVIEW OF RULEMAKING  
- CONTROL OF RULEMAKING  
- COST ANALYSIS OF REGULATIONS |
| - MANAGEMENT OF GENERIC ISSUE RESOLUTION | - PRIORITIZATION OF PROPOSED GENERIC ISSUES 
- GENERIC ISSUE TRACKING SYSTEM |
IAEA TECHNICAL COMMITTEE
ON
THERMAL REACTOR SAFETY RESEARCH

REACTOR SAFETY RESEARCH ACTIVITIES IN YUGOSLAVIA

Prepared by:

B. MAVKO
"J. STEFAN" INSTITUTE, LJUBLJANA

D. FERETIC
FACULTY OF ELECTRICAL ENGINEERING, ZAGREB

LJUBLJANA
JUNE 1987
1. Introduction

At this time Yugoslavia has one 632 MWe nuclear power plant of PWR design, located at Krško. NPP Krško, which is a two-loop plant, started power operation in 1981. In 1986 the Krško Nuclear Power Plant produced 3824.5 GWh electric power which is 1.2% above the planned production. Generator on-line operation was 6561 hrs. By May 1987 the NPP Krško has supplied 20 billion KWh to the grid.

Preparatory activities are continuing for the planned series of nuclear power plants to be built up to the year 2000. In connection with this Yugoslav Association of Electric Utilities (JUGEL) has ordered a study to determine the bases for safety criteria to be used for these nuclear power plants. The "Josef Stefan" Institute has coordinated the effort in which several Yugoslav institutions took part and the resulting final report was published in 1986.

Particular attention in the past year was placed to the development of regulations covering nuclear safety requirements for the future plants. New federal regulations on Criteria for Siting, Design, Commissioning, Operation and Decommissioning, on Safety Analyses Reports, on Requirements for Operating Personnel and on Safeguards were drafted and are expected to be passed later this year.

2. Safety Research Activities

In general, reactor safety activities related to:
- upgrading the safety of NPP Krško,
- developing capabilities for future units,
are predominantly conducted at the following research centers:

- "J.Stefan" Institute, Ljubljana,
- "B.Kidric" Institute, Beograd,
- "R.Boskovic" Institute, Zagreb and
- Faculty of Electrical Engineering, Zagreb,
- Faculty of Mechanical Engineering, Zagreb,
- "R.Koncar" Institute, Zagreb.

It is highly appreciated that all safety related activities taking place in Yugoslavia are kindly supported by IAEA through technical
assistance projects and research contracts. The Agency assistance in large extent contributed also to the different bilateral cooperations with other member states.

3. Post Chernobyl Accident Activities

Immediately after the occurrence of the accident (besides activities related to the radiation protection which were initiated following the first news and which have been going on since - the results were reported at specific IAEA meetings), steps were also taken to address the accident from the standpoint of nuclear safety. The safety characteristics of RBMK reactors were analyzed and probable events sequences and consequences were estimated.

With the CRAC-2 code the upper limit of the expected acute consequences was assessed for the whole body and thyroid doses, the source term being taken as the steam explosion scenario in the PWR assuming direct wind from the site to our country. Later, as more information about the radioactive cloud were available its effects on acute irradiation of thyroid, total bone marrow, lower large intestine, lungs and the whole body were studied in detail, again with the help of CRAC-2.

Currently a report is being prepared which will address recommendations for the steps to be taken to further improve the operational safety of our plant based on lessons learned in the passed year.

4. Probabilistic Safety Analysis

Yugoslavia has joined the Interregional IAEA project, "Probabilistic Safety Analysis" in late 1985. This project, locally coordinated by IJS, is performed jointly by different institutions from SR Slovenia and SR Croatia.

The analysis comprises the evaluation of importance and interaction of various plant systems and components, identification of accident sequences, deficiencies of design, of operating procedures and of test and maintenance work. The overall objective of this program is to produce a probabilistic model of the Krsko NPP which can be used by the utility and regulators in both safety and operational analysis of the plant. The project scope is a level 1 PSA covering four initiating events:

- small break loss of coolant accident
- steam-generator tube rupture
- steam line break
- station blackout.

The present goal is assessment of the core melt frequency for these accident initiators internal to the plant, including Loss of Offsite Power.

As already reported the assessment which started by reliability
analysis of various plant systems (MFWS, AFWS, CSS, E1, ECCS -HPI, RHR -, ACC, ESWS) is being continued by the development of event trees based on scenario already analyzed using deterministic approach. It is expected that by the end of this year this task will be completed and plans are in progress to expand at least level I assessment to a larger number of initiating events and later proceed with levels 2 and 3.

5. Deterministic Safety Analysis

At "J. Stefan" Institute and at the Faculty of Electrical Engineering in Zagreb which closely cooperates also with Faculty for Mechanical Engineering, R.Koncar Institute and R.Boskovic Institute all of them in Zagreb the reactor safety research based on deterministic approach is continuing.

The basic fields of research cover deterministic transients and accidents analysis, heat and mass transfer, two-phase coolant flow, dynamic behavior of systems and plant components during transients and structural analyses of components and constructions.

For these purposes, the intensive work started on the transfer of the RELAP5/MOD1 code from CDC and IBM environment to VAX 750. The work has progressed to the point when we expect to have RELAP5/MOD1 working on a VAX 750 computer by the end of this summer.

At IJS in Ljubljana a large part of deterministic analyses was directed towards the assessment of steam generator tube plugging effects on normal and transient operation of the Krsko NPP. A large break loss-of-coolant accident was recalculated for different extend of the plugged tubes using the conservative evaluation model and the computer code RELAP4/MOD6. The results of the preliminary analysis showed the relative increase of the maximum cladding temperature depending on the plugging rate. The results also showed the necessary peaking factor reduction to compensate for conditions caused by the plugging.

With the help of the RELAP4/MOD6 code the IJS and FEE analyzed the standard problem which was performed under the sponsorship of the Agency and is based on the PMK-NVH test facility in Hungary simulating SB LOCA on the VVER power plants. The analysis of the calculated results showed a good agreement with the measured data.

The IJS staff developed two new computer codes: SMUP-05 and MS-01. The former analyses the steady-state mode of the U-tube steam generator with the preheater of the Westinghouse-D4 type, while the latter is used to determine the steady-state model of the main steamline with turbine control valves. Both models were used to analyze and determine new operating conditions in the primary and the secondary system under various conditions resulting from modifications or parameter changes in the primary or in the secondary system. The model was also used to analyze the steam generators tube plugging effects on the parameters of the primary and the secondary side during normal operation.

To evaluate Krsko NPP core capability, a model was developed for
the design calculation of the reactor core thermal-hydraulics using the COBRA-III-C computer code. Departure from the minimum boiling ratio was analyzed depending on the design power history distributions, on the type of the core (KWU,W). In addition, the structural analysis of fuel rods was conducted with FRAPCON-2 and the thermal-mechanical response studied during the reactor operation. In this area IJS is in cooperation with the University of Erlangen, FRG, developing a code for determination of non-steady-state temperatures of fuel pellet axially-symmetric problems with fractures and melting by a self-adaptive mesh grid using the boundary element method.

With RELAP5/MOD1, cycle 19, steam generators tube plugging effects were studied for two very important assumed accidents:

a) small break loss-of-coolant,
b) rupture of one or more steam generator tubes.

The effects of 10% plugging were studied obtained for break sizes equivalent to 3/8, 1/2, 1", 2" and 3" and for one and 5 ruptured SG tubes. In addition last year IJS joined the international standard problem ISP-20 analyzing data from a steam generator tube rupture transient.

The IJS, Ljubljana and Faculty of Electric Engineering, Zagreb established also a close cooperation with Gesellschaft fuer Reaktorsicherheit in Garching. The ALMOD computer code which was transferred few years ago has been improved jointly, so that it is now capable of simulating non-symmetric phenomena in multi-loop plants. The modified ALMOD code was verified with the calculations of startup tests and with the results of measurements in the Krško NPP and used by FEE to calculate consequences of loss of all AC power and study the plant behavior in transients following a determined percentage of plugged tubes in case of turbine trip and in case of loss of feedwater.

7. Structural Analysis

The IJS staff have successfully transferred to VAX 750 computer the SAP-IV code, used for the analyses of various kinds of structures, components and pipes. It uses the finite element methods for static and dynamic structural analyses of elasticity. The program was supplemented by subroutines for space plotting of components and deformations. The first test calculations of the reactor vessel head vent and the three-dimensional model of a thick-wall pressure vessel of the nuclear reactor are showing promising results.
Commission of the European Communities
Joint Research Centre
Ispra Establishment

THE COMMISSION'S LWR-SAFETY
PROGRAMME 1984-1987

A. MARKOVINA

For presentation at the IAEA Technical
Committee on Thermal Reactor Safety Research
Vienna - 9/12 June 1987
0. Introduction

The Commission's LWR-Safety Programme includes research activities, which are performed directly at JRC and fully financed by the Commission (Direct Action - DA) and research activities which are supported by the Commission with financial contributions generally of abt. 50% and are executed in the EC national laboratories (Shared Cost Action - SCA).

The overall objectives of the combined DA and SCA in the LWR-Safety domain are:

- Accident prevention
- Accident analysis, control and accident consequence mitigation

In the area of accident prevention the main effort of the Commission is devoted to risk and reliability assessment and to studies of the integrity of components and systems.

In the area of accident analysis, control and accident consequence mitigation the effort is devoted to a deeper understanding of the basic phenomena occurring during the accidents and to the development and assessment of models and computer codes simulating the different phases of an accident.

The JRC LWR Safety Programme is subdivided into the following 5 research areas:

- Reliability and Risk Evaluation
- PISC (Project for Inspection of Steel Components)
- LWR Primary Circuit Components Life Prediction
- Study of abnormal behaviour of LWR Cooling Systems
- Source Term

An additional area was included in 1987 named: Containment studies.
RESEARCH AREA 1: Reliability and Risk Evaluation

The objectives of this research area are to develop and harmonize Probabilistic Risk Assessment (PRA) methodologies, to create a viable information system on reactor operation in order to provide the necessary feedback to decision makers (from reactor operators to licensing authorities), to study and develop tools able to simulate and control various accident sequences.

The following three main activities are carried out:

- Implementation of a European Reliability Data System (ERDS) to collect and organize data on the operational history of nuclear power plants in Europe.

The system is composed of the following data banks:

- Component event data bank (CEDB) dealing with main component failures
- Abnormal Occurrence Reporting System (AORS) dealing with incidents and accidents
- Operating Unit Status Report (OUSR), dealing with plant outages.

- Assessment of procedure and methods of Probabilistic Risk Analysis (PRA)

Following the successful completion of the System Reliability Benchmark Exercise (SBE), centered on the assessment of common cause/common mode failures of the feedwater system of a reference European NPP, other SBEs are planned. These SBEs have proven to be particularly amenable to be carried out as Shared Cost Action, providing an elevated benefit/cost ratio.

The Human Factor SBE is planned for the period Sept. 86 - Sept. 87 and will be followed by the Event Sequence SBE of about the same duration.

- Implementation of the System Response Analyzer

Aiming to explore the feasibility and to develop techniques and models that would make possible the simulation (in real time or faster) of accident sequences, including the simulation of operator and system behaviour.

RESEARCH AREA 2: PIISC (Project for the Inspection of Steel Components)

The objectives of this research area are to assess and improve Non Destructive Testing (NDT) techniques and related models which contribute to the safety assessment of pressurized steel components. In particular the aim is to verify the effectiveness of present inspection procedures, to identify shortcomings of techniques, to improve the inspection of austenitic steel components and equipment knowledge.
Non Destructive Testing (NDT) techniques are used for the in-service inspection (ISI) of a reactor pressure vessel (RPV) and its associated pressure circuit, and the results of the NDT are used to assess the significance of flaws. Detection, location and sizing of crack like defects play an important role in helping to establish the integrity of reactor steel structures.

There is a need to have information on the accuracy of NDT for sizing and locating flaws and on the reliability with which flaws are detected.

The first trial to give quantitative evidence of the ability of ultrasonic testing (UT) to find and size crack like flaws was the PZSC I round robin test (RRT).

A second set of round robin trials to collect more data was the PISC I programme sponsored jointly by the Nuclear Energy Agency (OECD-NEA) and the Commission of the European Communities. It was also considered necessary to add separate studies on certain parameters which influence flaws detection and sizing, to assist both in understanding the dispersion in the RRT results and in assessing the performance of the specific techniques applied.

The studies of two of these parameters, namely the defect characteristics and the cladding characteristics, have been included in the SCA programme.

A third study, also included in the SCA programme, concerns the effect of real geometries and real working conditions. In particular for this study a full scale vessel of a BWR and a ring with PWR components are made available, with all the infrastructures to perform in-situ inspections.

RESEARCH AREA 3: LWR primary circuit components life prediction

The objectives of this research area are to develop and validate techniques and models for the quantitative assessment of the reliability of reactor pressure vessels and pipings. The attention is particularly focused on the estimation of the residual lifetime of primary circuit components.

During the 1980-83 programme the JRC has set up a laboratory for fatigue loading and monitoring by means of various standard and advanced Non Destructive Testing techniques.

Three scale 1/5 models of PWR vessels have been manufactured and equipped with a scanning facility for internal automatic inspection by ultrasounds. In parallel the development has been started of a code (taking COVASTOL code as starting point) for the continuous life time prediction of Reactor Pressure Vessels.
In the 1984-85 period the 1/5 RPV models are submitted to mechanical fatigue at room temperature (by cyclic hydraulic pressurizations), with the objectives of testing a methodology for residual life prediction and reliability assessment and of applying both advanced and classical NDI techniques for fatigue crack growth detection and sizing.

Two SCA activities, tightly connected to be above objectives, are:

1. Structural Reliability Benchmark Exercise (SRBE)
2. Non destructive inspections (NDI) on the JRC-Ispra vessels

The first activity consists of a cross comparison of different predictions of vessel integrity deterioration during fatigue cycling and of a comparison of them with experimental results coming from the JRC 1/5-scale vessels.

The second activity aims at formulating an optimized inspection procedure (continuous and periodic), suggesting techniques and methodologies for the detection of fatigue crack propagation.

RESEARCH AREA No 4: Study of abnormal behaviour of LWR cooling systems

The overall objective of this research area is to develop validated computer tools for the analysis of abnormal thermohydraulic conditions in LWRs as created in loss-of-coolant accidents (LOCA) due to breaks in the primary containment or in special transients.

The activities carried out in this area are:

- the generation of experimental data with the LOBI-MOD2 integral facility in view of extending and/or complementing the existing data base for LWR safety related thermohydraulics;
- the analysis of these data in view of a deeper understanding of the thermohydraulic mechanisms and phenomena;
- the use of these data for the development and improvement of analytical models and for the independent assessment of large system codes applied in water reactor safety analysis.

A special analytical effort is devoted to the assessment and improvement of large system codes developed and utilised in the Community, like CATHARK, DRUFLAN and the USNRC code RELAP 5.

This programme is complemented by a series of SCA activities.

The first one is a set of review studies on the state-of-the-art of separate effects and component behaviour with emphasis on SB-LOCA conditions, concerning the following phenomena:
- contact condensation effects relevant to ECC water injection into a cold leg main coolant pipe
- two-phase flows in branches of a horizontal pipe
- counter-current flow limitations (CCFL) in horizontal and vertical pipes
- hydraulic behaviour of a partially uncovered core.

The second one is an experimental investigation and modeling of heat transfer in the transition region, including the minimum film boiling point.

The third one concerns the participation of national organizations in system code assessment. This activity is devoted in particular to the assessment of CATHARE, DRUFAN and RELAP5/MOD2 with the LOBI experimental data. A test matrix was agreed among the participants to make complementary analysis and cross-check exercises.

The fourth one is the analysis of some experimental results produced in the previous SCA-Reactor Safety 1980-83 programme. It consists of a set of analytical studies performed, also in this case, with computer codes largely utilized in Europe, such as TRAC FF1/MOD1, CATHARE and RELAP5/MOD2. The following four topics are the subject of this analytical work:

- Revetting propagation over Zircaloy under bottom reflooding conditions
- Two-dimensional effects in the core at a PWR during the reflooding phase of a LOCA
- Fluid dynamic effects in the fuel element top nozzle area during refilling and reflooding
- Heat transfer to a dispersed two-phase flow and detailed quench velocity research.

RESEARCH AREA 5: Source Term

The overall objective of this research area is to achieve a consensus view in the European scientific community on the mechanistic modelling of fission product and aerosol and behaviour in severe accident scenarios and to make available to all EC member countries experimental results and analytical tools.

The main effort of the Direct Action programme is addressed to the studies of transport, deposition and retention processes taking place in the reactor containment building, in particular of the type "large dry PWR". The basic codes being utilized and improved in these studies are CONTAIN and MARCH 2. As a part of this programme, JRC participates in the LACE project, sponsored by EPRI (U.S.) and financed by an international consortium.

The Shared Cost Action programme is concentrated on experimental activities dealing with the problem of aerosol and fission product resuspension in the reactor containment building and to the chemical transformations induced by
specific events, such as high temperature due to $H_2$ combustion. The first one of these activities concerns the resuspension processes due to a relatively mild depressurization of the containment atmosphere. The second one deals with more dynamic effects, involving rapid changes of flow regimes and temperatures.

RESEARCH AREA: Containment studies

This research area has been included in the JRC Reactor Safety programme in 1987. The objective is to assess the status of knowledge of the containment building structural behaviour in the non-linear domain of deformation and the need of an experimental activity focused on the utilization of the Large Dynamic Test Facility and other dynamic loading facilities available at JRC-Ispra.

The activity is primarily addressed to concrete structures, either prestressed or reinforced, with and without an internal steel liner. In the frame of the SCA review studies are undertaken to evaluate the capabilities of the presently available analytical tools to predict the mode and timing of containment leakage and rupture under the severe loads included in the design and design verification stage (e.g. seismic loads) and under the loads originated by severe accidents (e.g. $H_2$ detonation/deflagration).

As part of the DA, scoping tests on reinforced concrete specimens are carried out to assess the effect of strain rate.
1. IWR SAFETY PROJECT INDEX

1.1 Background

The IWR Safety Research Index has been published by CEC for many years as a compilation of information on research projects relating to IWR nuclear safety. Since 1981, it has been published biannually. The number of contributions has steadily increased and reached the level of 1700 pages, in which more than 600 project descriptions have been collected from 10 Member countries.

In parallel to this activity, the OECD/NEA has also compiled a similar Index for its member countries alternating with the CEC publication. Furthermore, in 1982 the IAEA started a trial collection of information on limited areas of particular interest and has obtained a number of contributions from its member countries including non-OECD countries.

Computerisation of the Index has been discussed in some form at the CEC-XII/D/1 since 1974 in the CSNI/NEA and IAEA.

1.2 ADVANTAGES RESULTING FROM THE COMPUTERISATION OF THE NSRI

Although the manpower and costs required for the initial input of the entire contents can be an obstacle to the computerisation of the Index, the following advantages have been identified:

(a) Searching by Keywords

It will become possible to search a preselected set of subjects throughout the information stored on a computer.

(b) Easy Updating of the Project Descriptions

The contributors could easily update their respective project descriptions just by indicating corrections to the old edition of the Index and by adding descriptions of new projects. This would also enable more frequent updating of the Index.

(c) Compilation of Various Documents

Since it would be relatively easy to reclassify the project descriptions according to any specific order, other documents on related subjects can be compiled, such as annual reports on the final results of the completed research projects.
during the year. Thus the information contained in the Index could be utilised for a variety of purposes.

1.3 **MEETINGS WITH OECD AND IAEA REPRESENTATIVES**

Two meetings were held: the first one in Paris (OECD/CSNI) on the 11.6.1986 with representatives of IAEA, OECD and CEC.

**Participants:**

Messrs. M. JANOWSKI (IAEA)
J. MORIMOTO (OECD)
R. PRIMAVERA (CEC/ISPRA)
E. DELLA LOGGIA (CEC-XII/D/1)

During the meeting it was reviewed the current status of the Nuclear Safety Research Index (NSRI), its FORMAT, STRUCTURE, the Number and Quality of contributions.

It was also discussed the possibility of jointly publishing the NSRI, under the sponsorship of the CEC/OECD/IAEA. It was proposed the following arrangement for the next three years:

i) OECD will prepare the 1986 edition of the Index (limited, as until now, to OECD countries);

ii) CEC will prepare the edition of 1987 (limited to CEC countries);

iii) For the 1988 edition, it is proposed to joint Index IAEA/OECD/CEC, computerized;

iv) IAEA will, with the cooperation of CEC and OECD, prepare the system for the computerization of the Index;

v) IAEA will collect the entries also for the 1987, as preparation for the 1988 joint Index.

They are prepared to offer their entries to CEC for inclusion on our INDEX for 1987, if we agree.

On the second one, in Brussels on the 30.10.1986 at CEC headquarters the participants were:

Messrs. M. JANOWSKI (IAEA)
P. CAPOBIANCHI (CEC/ISPRA)
R. PRIMAVERA (CEC/ISPRA)
E. DELLA LOGGIA (CEC-XII/D/1)
1.4 THE OUTCOME OF THE MEETING OF THE 30.10.1986

Messrs. CAPOBIANCHI and PRIMAVERA presented the main features of MARS (Major Accident Report System), they propose to adapt to the joint IAEA/OECD-NEA/CCE Index.

The discussion touched upon the requirements of the computerised system for the Index such as entries, classification, controls, keywords, output, etc.

It is recommended to pass a small contract to an outside firm (of the order of 10.000 - 15.000 ECU) for the adaptation of MARS to the Index (to the same firm which is working on MARS). A meeting should be organised to discuss with IAEA and OECD-NEA representatives the proposed system and the organisation of the work.

The system requires a qualified person to entry in the data bank the formats, to adjourn the entries and to keep the contacts among the 3 partners.

1.5 PROPOSED PLAN OF WORK

a) Meeting with IAEA and OECD-NEA to agree on the technical details of the proposed system.

b) Contract for the adaptation of MARS to the joint Index system

c) Meeting in Vienna organised by IAEA to discuss with the national representatives the format of the entries, the organisation of the data collection, etc.

d) End of 1987 - Preparation of Index by CEC, possibly computerised.

IAEA will collect the entries at the end of 1987 as preparation for the 1988 Joint Index; they are prepared to offer their entries to CEC for inclusion in our computerised system.

e) End of 1988 - Joint computerised Index by IAEA/OECD-NEA/CCE.

2. JOINT CEC-IAEA STATE-OF-THE-ART REPORT ON HYDROGEN GENERATION, DISTRIBUTION AND EXPLOSION POTENTIAL ASSOCIATED WITH ACCIDENTS IN LAR

Following on the TMI 2 accident a CEC experts subgroup was formed whose prime task was to assess the situation on Hydrogen generation, distribution and explosion potential associated with accidents in LAR research both inside and outside Europe and to recommend further research work to be performed in European laboratories. A State-of-the-Art Report was prepared by the subgroup which was issued in 1981 and updated in September 1985. It includes a review of the objectives and results of various international investigation programmes. This is now being incorporated in a joint CEC-IAEA State-of-the-Art Report which is
due to be published soon.

Members of the CEC subgroup have been actively involved with the production of the collaborative report.

3. ACTIVITIES ON SOURCE TERM BY CEC – BRUSSELS.

3.1 The following three studies have been completed, namely

a) "STEAM CONDENSATION MODELLING IN AEROSOL CODES"

The physical modelling of condensation, the mathematics of the discretisation of the equations, and the methods for modelling the separate behaviour of different chemical components of the aerosol have been addressed.

b) "COMPARISON OF EUROPEAN COMPUTER CODES RELATIVE TO THE AEROSOL BEHAVIOUR IN PWR CONTAINMENT BUILDINGS DURING SEVERE CORE DAMAGE ACCIDENTS"

The study concerns a comparative exercise of computer codes used to assess the aerosol behaviour in PWR containment buildings during severe core damage accidents (with steam condensation onto the particles).

c) "COMPARISON OF AEROSOL CODE RESULTS WITH DEMONA B3 EXPERIMENTAL RESULTS".

The study shows the results of an exercise of comparison between calculated values and results from a DEMONA experiments.

It is worthwhile to mention a Study aiming at the preparation of a thermochemical data book for reactor materials and fission products. The first two parts of the report have been already prepared; for the final one a Study Contract is being prepared. The objective of the study is to present sets of critically assessed thermodynamic data for reactor materials and compounds of fission product elements. The elements chosen are those which, if released to the environment, would present the most significant radiological hazard. The data provided are relevant to both water and fast breeder reactors.

3.2 Two Study Contracts have been allocated:

1) "STATUS REPORT ON PHYSICAL AND NUMERICAL THERMAL-HYDRAULIC CONTAINMENT ANALYSES CAPABILITIES WITH REGARD TO T-H/AEROSOL COUPLING".

The study aims at summarising the existing know-how in thermal-hydraulic containment analysis in the format of a status report. In the frame of this Study a Thermo-hydraulics experts subgroup from Member Countries meet regularly to discuss the most relevant topics in this field.
2) "DIFFUSIOPHORESIS OF FISSION PRODUCT AEROSOL IN A LIAR CONTAINMENT AFTER CORE MELTDOWN".

ECN (NL) has performed an extensive experimental programme on the subject. The study will allow to complete the interpretation of the experiments and to discuss the proposed model for the diffusio-phoresis.

Concerning Chernobyl accident, an assessment has been prepared by the services of the Commission; the report at the moment has only internal distribution. This assessment was summarised in a report to the European Parliament.

The Commission has also allocated a study contract with the aim of providing an "ASSESSMENT OF THE CHERNOBYL ACCIDENT DYNAMICS OF RELEVANCE FOR SOURCE TERM EVALUATION".

Consideration is given to those effects and phenomena of the accident sequence which mostly determined and influenced the radioactivity release.

4. RADIPROTECTION

Within the frame of the Radiation Protection Research Programmes, ten post-Chernobyl activities will be launched.

These are dealing with problems of environmental contamination countermeasures and the assessment of health consequences.

Several of these activities are re-enforcement of the Commission’s ongoing MAREA project on the evaluation of the consequences of accidental releases of radioactive materials.

5. COMPUTER ORIENTED TECHNIQUES AND ASSESSMENT OF ENVIRONMENTAL CONSEQUENCES WITHIN EMERGENCY PROCEDURES FOR NUCLEAR ACCIDENTS.

It had become evident by 1985 that different member states of the EEC are introducing the use of computers into emergency procedures in very different ways. Hence a review of the current situation was carried out by an international study group of experts from EEC and Nordic countries to suggest where future collaboration and coordination could be of benefit. A draft report is now being considered by the EEC.

A workshop organised by the European Commission in Luxembourg in September 1985 and an IAEA symposium in Rome in November 1985 had already provided useful input. More recently the Chernobyl accident has emphasized the lack of coordination and international standards for data acquisition and nuclear accident consequence assessment.

Several nuclear power plants currently in operation and planned in Europe are within 20 to 50 km of national boundaries. Collaboration in compatible development of computer use in this...
context is therefore highly beneficial.

The discussions contributing to this report centred almost exclusively on conventional power reactors in Western Europe. Reconsideration of emergency procedure is under way for other nuclear installations.

The study group recommends establishing means of international cooperation in development and harmonization in the use of computerised assistance for emergency situation.

6. CONTAINMENT INTEGRITY AND LEAK TESTING.

6.1 In the framework of the activities of Working Group No.1 on safety of light water cooled reactors a report was prepared and finished in November 1986 on leak testing procedures applied and experiences gained in European countries.

The conclusions of the report are that although leak testing methods have achieved a rather satisfactory state there are still divergent views concerning specific phenomena on which the quantitative assessment depends.

Divergencies also still exist concerning the allowable leakage rates, the pre-operational and periodic test pressure and the test intervals.

6.2 Structural containment design to withstand accident situations.

A state-of-the-art report is under preparation (and we expect its completion before the end of 1987) which describes and compares in detail the national rules and practice applied within the member countries of the European Communities and in the United States of America. Specific emphasis will be put on the application of codes to determine local and global design loads, the structural design codes and the resulting inherent margins of conservatism when rules and guidelines are thus exercised.
STEERING COMMITTEE FOR NUCLEAR ENERGY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

A GLANCE AT
NEA'S PROGRAMME OF WORK
IN THE NUCLEAR SAFETY AREA

(Spring 1987)
In 1948, the United States offered Marshall Plan aid to Europe, provided the war-torn European countries worked together for their own recovery. This they did in the Organisation for European Economic Co-operation (OEEC).

In 1960, Europe's fortunes had been restored; her standard of living was higher than ever before. On both sides of the Atlantic the interdependence of the industrialised countries of the Western World was now widely recognised. Canada and the United States joined the European countries of the OEEC to create a new organisation, the Organisation for Economic Co-operation and Development. The Convention establishing the OECD was signed in Paris on 14th December 1960.

Pursuant to article 1 of the Convention, which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and this to contribute to the development of the world economy;

- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and

- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Signatories of the Convention were Austria, Belgium, Canada, Denmark, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries acceded subsequently to the Convention (the dates are those on which the instruments of accession were deposited): Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971) and New Zealand (29th May 1973).


**N E A**

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA, established on 20th December 1957) on the adhesion of Japan as a full member.

NEA now groups all the European Member countries of OECD and Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.
The primary objectives of NEA are to promote co-operation between its Member governments on the safety and regulatory aspects of nuclear development, and on assessing the future role of nuclear energy as a contributor to economic progress.

This is achieved by:

- encouraging harmonisation of governments' regulatory policies and practices in the nuclear field, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;

- keeping under review the technical and economic characteristics of nuclear power growth and of the nuclear fuel cycle, and assessing demand and supply for the different phases of the nuclear fuel cycle and the potential future contribution of nuclear power to overall energy demand;

- developing exchanges of scientific and technical information on nuclear energy, particularly through participation in common services;

- setting up international research and development programmes and undertakings jointly organised and operated by OECD countries.

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other International organisations in the nuclear field.

An outline of NEA's main activities is given in Annex III.

**CSNI**

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing. The Committee was set up in 1973 to develop and co-ordinate the Nuclear Energy Agency's work in nuclear safety matters, replacing the former Committee on Reactor Safety Technology (CREST) with its more limited scope.

The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries. This is done in a number of ways. Full use is made of the traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences. Some of these arrangements are of immediate benefit to Member countries, for example by improving the data base available to national regulatory authorities and to the scientific community at large. Other questions may be taken up by the Committee itself with the aim of achieving an international consensus wherever possible. The traditional approach to co-operation is reinforced by the creation of co-operative (international) research projects, such as PISC, LOFT and the TMI-2 Sample Examination Programme, and by the organisation of international standard problem sessions.
exercises, for testing the performance of computer codes, test methods, etc. used in safety assessments. These exercises are now being conducted in most sectors of the nuclear safety programme.

The greater part of the CSNI co-operative programme is concerned with safety technology for water reactors. The principal areas covered are operating experience and the human factor, reactor system response during abnormal transients and accidents, various aspects of primary circuit integrity, the phenomenology of radioactive releases in reactor accidents, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

The Sub-Committee on Licensing, consisting of the CSNI Delegates who have responsibilities for the licensing of nuclear installations, examines a variety of nuclear regulatory problems and provides a forum for the review of regulatory questions, the aim being to develop consensus positions in specific areas.

**NUCLEAR SAFETY TECHNOLOGY AND LICENSING**

**HIGHLIGHTS**

NEA's programme has always included substantial work related to the safety of nuclear power. Originally focussed only on radiation protection aspects, this developed in the 1960's and early 1970's to include a large number of questions in the field of reactor safety technology. With more power reactors coming into service and the increasing emphasis on health and safety, OECD countries became more concerned with the problems of licensing their nuclear installations and operating them in a safe and reliable manner. Over the years the NEA has taken care to continually adapt its working structures to meet these needs.

With respect to reactor safety research the Committee on the Safety of Nuclear Installations has identified seven main areas as worthy of specific international co-operation at this point in time. These areas are: operating experience and human factors, reactor transients and loss-of-coolant accidents (LOCA), primary circuit integrity, reactor accident source terms and accident consequences, risk assessment, severe accidents, and containment performance. Exchanges are also taking place on the topic of the safety of the nuclear fuel cycle. The highlights of the work in these different areas are described in the following pages.

At present, CSNI work is focussed on the safety of thermal reactors, primarily on safety aspects of light-water reactors. Other types of reactors (thermal as well as fast reactors) and nuclear fuel cycle facilities are being considered according to the needs and the requests of the Member countries.

During 1986, the nuclear safety scene was dominated by the Chernobyl accident. A large fraction of NEA's programme of work was devoted to an assessment of the relevance of Chernobyl to the safety of commercial nuclear power plants operated in OECD Member countries.
LICENSING

Major issues relating to the licensing of nuclear installations, such as important regulatory measures adopted by Member countries and major new departures in national regulatory policies, are discussed by the CSNI Sub-Committee on Licensing. With around 300 power reactors now in service throughout the OECD area and another 75 under construction or at the planning stage, it is no longer possible for research issues and licensing questions to be seen as separate. Many regulatory requirements are intimately bound up with technical problems dealt with in the Agency's nuclear safety programme. The Sub-Committee on Licensing offers an appropriate forum for discussing the impact of the latest scientific and technical developments on regulatory requirements and for exchanging views on regulatory issues or measures before decisions are taken. Regarding the effect of the Chernobyl accident on nuclear reactor regulations and licensing procedures, the Sub-Committee for Licensing, at its 1986 meeting, agreed that the accident did not necessitate any immediate changes in the regulatory positions in Member countries but the Sub-Committee would continue its review of further information about the accident as and when it becomes available.

Special Meetings of the Sub-Committee on Licensing are also held annually to discuss subjects of topical interest. A meeting to discuss trends regarding source terms and severe accidents was held in 1986. In 1987, a meeting is planned on regulatory approaches for the consideration of reactor containment performance beyond the design basis.

INFORMATION ON SAFETY RESEARCH

International collaboration on reactor safety depends heavily on the availability of reliable, detailed information on Member countries' safety research programmes. A fundamental source of such information is provided by the Nuclear Safety Research Index, which has been published regularly by the NEA since 1971, and is now issued every two years jointly with the OECD International Energy Agency. The 1984 Index compiled standard details on some 900 current research projects relating to the safety of water reactors in Member countries, including government bodies, public and private research institutes and universities. The 1986 edition of the Index will be published during the spring of 1987. In the future the compilation and publication of this document will be carried out alternatively by the NEA/IEA and the IAEA.*

PRIORITIES IN LWR SAFETY RESEARCH

The shift in emphasis from construction to operation of nuclear power plants, plus the fact that the nuclear power industry had reached a certain degree of maturity, has led to reduced public financial support, in some countries, for nuclear development and in some cases even for nuclear safety research. Coupled with the effects of budgetary stringencies, these reductions affected the capability of the government agencies concerned with public health and safety to respond fully to safety problems that are certainly likely to continue to arise for several years to come. In view of

* The IAEA will produce the 1988 Index.
this the CSNI set up an ad hoc group to compare national insights and attitudes and thus determine the priority areas for research. It was found to be possible to arrive at essentially converging views concerning the more prominent areas of nuclear safety research but of course the existence of certain special circumstances in individual countries (such as the maturity of the nuclear industry in that country) make the recommendations not universally valid in all cases.

OPERATING EXPERIENCE AND HUMAN FACTORS

Incident Reporting System

The NEA Incident Reporting System (IRS) was set up in 1980 to collect and disseminate information on safety-related operating experience in nuclear power plants. It serves two main purposes. Firstly, it enables regulatory authorities and electric power companies to take the necessary action to prevent reported mishaps from recurring. Secondly, it helps in identifying areas where design or operation of plants can be improved. The system has now been in operation for more than six years, with active participation by all of the 13 OECD Member countries that have nuclear power plants in service. By the end of 1986, more than 830 reports corresponding to more than 800 incidents had been circulated within this framework.

It is to be noted that there were no incidents in nuclear power plants in OECD countries which caused significant radiological consequences to the health and safety of the general public. In order to maintain this favourable record with the growing number of plants, a continuing effort is required to make the best use of the accumulated information. As a measure to facilitate a better feedback of lessons learned from incidents, a Guidance for the Preparation of IRS Reports has been developed, with a view to improving the quality of IRS reports as regards consistency and completeness.

In 1984 co-operation with a similar incident reporting system set up by the IAEA was initiated to explore the possibility of world-wide coverage. In accordance with the agreement between the agencies, NEA acts as the clearinghouse among the OECD countries for exchanging information between OECD and non-OECD countries. A selection of incident reports contributed by OECD countries were forwarded to the IAEA, while those from non-OECD countries were received in return for distribution within the OECD countries. Currently reports are being exchanged every half-year on a batch basis, but the frequency of the exchange of reports is being increased in order to strengthen co-operation between the NEA-IRS and the IAEA-IRS.

In the same context, a fourth meeting for the exchange of information on incidents was held under joint sponsorship with the IAEA. Important areas of safety concern were highlighted, including technical problems with main steam isolation valves and wear of instrumentation thimbles; incidents occurring under specific operating conditions; effect of external events and cumulative occurrence of independent events and system interaction. The discussion also led to general remarks that human factors were an influence in most situations and that preventive maintenance and periodic testing were very important.

In order to assist Member countries in retrieving the rapidly
increasing amount of information exchanged through the NEA-IRS, an IRS Data Bank has been established in co-operation with the Joint Research Centre of the CEC. Interested Member countries are making use of the information stored in the Data Bank either in the form of a magnetic tape or by interrogation through the international telephone network. As a follow-up to the Chernobyl accident and in order to rapidly develop an improved computerised data base on incidents, the data base structure and coding guidelines of IRS are being improved and the incident reports, including the backlog of standardised ones, are newly input into the NEA Data Bank.

It is foreseen that the NEA Data Bank will play an increasing role in the IRS data handling, especially in order to broaden and deepen the incident analyses.

**Symposium on Reducing Reactor Scram Frequency**

Recent operating experience with nuclear power plants shows that there are wide differences in the frequency of reactor scrams between plants in different countries. NEA Member countries are interested in the possibility of improving safe and reliable plant operation by reducing scram frequencies. A first step towards this goal is gaining a clear understanding of the underlying reasons for the differences in the observed scram frequencies, with a view to devising appropriate remedial measures.

Accordingly, a Symposium on this subject was convened in Tokyo, Japan in April 1986. The purposes of the Symposium were to compile consistent statistical data regarding scram frequencies in OECD Member countries; to exchange insights into the reasons for the differences between countries; and to look for measures by which plant performance can be improved.

The discussions covered wide-ranging aspects, such as operating experience, features and procedures provided to reduce reactor scram frequency, policies on maintenance and testing.

A consensus on the need to further reduce unnecessary scrams was confirmed, and as a follow-up to the recommendation at this Symposium, data on aggregated scram frequencies for respective plant types are being collected on an annual basis.

**Human factors**

Analysis of incidents in nuclear power plants has clearly underlined the need to focus closer attention on the importance of human factors in reactor operations. In fact a significant portion of incidents have been caused or aggravated by human error during operations under normal or accidental conditions, as well as in the course of maintenance and testing activities.

The Chernobyl and TMI accidents highlighted the considerable potential for human interactions to affect the safe operation of nuclear plant. Therefore, in addition to ongoing activities: Analysis of Incidents Involving Human Factors; Training Programmes for Plant Personnel and Use of Digital Computers in Control Rooms, a new Human Factor activity, Misinterpretation of Plant Status by Operators, will be highlighted. The objective is to understand the occurrence of incidents, or situations where the operator does not fully appreciate the state of the plant or the likely consequences of his
actions. The Task Force will be established and commence activities in 1987.

Since 1984, attempts have been initiated to collect information on research activities relating to human factors currently in progress in Member countries. The contributed project descriptions which were compiled as the "Newsletter on Human Factors", covered such aspects as human reliability qualifications, etc. The wide-ranging interest from various organisations, including government bodies, research laboratories and electrical utilities, led to a consensus that the information should be updated on a regular basis.

**REACTOR TRANSIENTS AND LOSS-OF-COOLANT ACCIDENTS**

Any assessment of nuclear power plant transients depends heavily on an understanding of the thermal and hydraulic phenomena taking place in the reactor cooling system. Clearly, a purely experimental approach to assessing plant response to such transients is not feasible, and in its place numerous and complex computer programs have been developed to calculate power plant response to different transients - and hence accident sequences - for different power plant designs. The results of these calculations provide the basis for decisions about the design of emergency core cooling systems (ECCS), and for the safety analysis of NPPs.

During development of these computer programs, their models and correlations have been related to experiments on individual phenomena and/or components (known as separate effects tests) and the behaviour of complete systems (known as integral tests), conducted in different facilities built to various scales. Comparison of code predictions with both types of experiments aims at the verification of a proper modelling of individual phenomena and components, and of a correct handling of components and systems interactions. It is also necessary to confirm that the computer programs can correctly extrapolate the phenomena observed in experiments to the full scale power plant system.

A major effort under the auspices of CSNI has been the development of a computer code validation matrix with the aim to provide a consistent basis by which the large computer codes used to analyse nuclear reactor cooling system behaviour under off-normal and accident conditions could be evaluated. The result is a matrix of experimental data and power reactor operating data which covers virtually all possible transients and accidents. In 1987 the matrix will be published as a CSNI report covering PWRs of both the U-tube and Once-Through Steam Generator designs and BWRs. In order to have a uniform basis, or criteria, on which code selection, validation and application can be made, in addition to the validation matrix a code assessment process is being developed which will have to be run through with each individual code in order to reach its assessment closure. The development of this assessment process and the preparation of a report describing the latter will be finalized by late 1987 or 1988.

A great amount of research has been performed worldwide aiming at the experimental and analytical investigation of the effectiveness of emergency core cooling systems (ECCS) and of the behaviour of nuclear reactor fuel under transient and accident conditions. The majority of the information exists as individual reports and publications. A report on the state-of-the-art of PWR fuel behaviour has been published in December 1986 as CSNI Report No. 129.
state-of-the-art report on ECCS effectiveness is under preparation and will become available by October 1989.

International research in the area of Boiling Water Reactor (BWR) containment response to accidents has been essentially completed; a state-of-the-art report on BWR pressure suppression systems has been published in October 1986 as CSNI Report No. 126. As an extension of this work, a state-of-the-art report on PWR containment is going to be prepared and will be ready by mid 1988.

A major part of the work on transients and breaks is selecting experimental programmes to be used for International Standard Problem (ISP) exercises. These exercises permit the participants to compare computer codes against each other and with experiments, and to identify the codes' effects. Two such exercises were concluded in 1985: ISP-16(HDR) and ISP-18(LOBI). One exercise was begun in 1985 and completed in 1986: ISP-19 (PHEBUS). Two new ISP exercises, ISP-20(Doel 2) and ISP-21(Piper I), one of which (ISP-20) will for the first time involve calculations for a full-scale PWR, will be performed in 1987. Further three tests, one from the Italian SPES, one from the French BETHSY and one from the Japanese ROSA-IV facility, are considered potential candidates for future ISP exercises in 1987 and 1988. The proposal for an ISP exercise on a counterpart test from all four integral system test facilities of different scale each, LOBI, BETHSY, SPES and ROSA-IV, is at present being discussed.

The work in the area of transients and loss-of-coolant accidents has begun to move closer to the severe accident conditions. This shift from the Design Basis Accident through the transition region into the Severe Accident is a natural evolution. Studies in conjunction with those specialising in Severe Accidents and the Source Term have begun in 1986. Joint efforts aim at analysing core debris material from Three Mile Island 2 and benchmark calculations of that accident. The initial view of the accident is that it can be divided into several phases: The design basis accident, the transition to severe core damage, and the result of the core melt. These phases will challenge the ability to link the analysis codes for reactor thermal-hydraulics/fuel damage/fission product release and transport. The work has begun in 1986 and will extend through 1987 and 1988, and possibly 1989.

OECD LOFT PROJECT

The Loss-of-Fluid Test (LOFT) facility is a small, 50MW(th), pressurised water test reactor designed to provide information on reactor system response during abnormal events or accidents. The facility is complete with all major reactor coolant system components and subsystems that are present in large commercial power reactors. The LOFT facility is unique in that it is the only thermal-hydraulic integral system test facility in the world to use nuclear fuel instead of electrically powered heaters. LOFT began nuclear operation in 1978 at the Idaho National Engineering Laboratory for the USNRC.

In 1983, NEA formed the international OECD LOFT Project to fund testing at LOFT of interest to participating NEA members. The facility is particularly well suited for experiments and acquisition of data on operational transients and multiple failure events, including fission product releases, that may
occur in a commercial reactor. Its versatility provides an excellent means for assessing and developing techniques for managing accidents.

The OECD LOFT Project membership consists of Austria, Finland, the Federal Republic of Germany, Italy, Japan, Spain, Sweden, Switzerland, the United Kingdom and the United States. The nuclear utility industries in the Federal Republic of Germany, Japan, the United Kingdom and the United States participate in the Project through Associate Membership.

The final LOFT test was performed during 1985. The test was the second of two planned fission product tests. As planned, a small break loss-of-coolant accident was initiated with the emergency core coolant system inhibited. This allowed the centre fuel module in the reactor core to heat up to the point of failure of the cladding material and relocation of the fuel. Of the fission products thus released into the reactor coolant system, the quantity, chemical composition and dispersion throughout the system were measured. Due to the large number of fuel rods in the LOFT core, the data obtained for the high rod temperatures and for the resultant damage are more typical than those from other smaller experiments and will therefore add considerably to the knowledge of damage patterns and progression in a full PWR fuel bundle. In addition, the information will be used in support of the programmes around the world that are re-evaluating the radioactive source terms used in predicting environmental consequences to large-scale reactor plant accidents.

The three year period of the OECD LOFT Project Agreement terminated in September 1986. To cover the comprehensive and detailed examination work on the centre fuel bundle as well as the thermal-hydraulic and fission product analysis work for this last test, FP-2, the previous Agreement was amended and resulted in the OECD LOFT Extended Programme which will run up to September 1989.

OECD HALDEN PROJECT

The Halden Reactor Project began in 1958 as an OECD co-ordinated nuclear fuel test programme using the 20MW(th) Halden boiling heavy water reactor in Norway. The tenth and present agreement extends the Project to 1987. The signatories to the present agreement include Denmark, Finland, the Federal Republic of Germany, Italy, Japan, the Netherlands, Norway, Sweden, the United Kingdom and the United States. Three industrial parties in the United States, that is, Combustion Engineering Inc., the Electric Power Research Institute, and General Electric Co., are associate members of the Project. Negotiations for an eleventh agreement covering the period 1988-90 are at present under way.

The Halden Project has expanded over the years to include a group studying the man-machine interface. This group has done pioneering work in the development of control room systems and displays to better help reactor operators understand the status of their power plant. The effort in this area is continuously expanding. The man-machine communication programme proposed for 1987 covers mainly three areas: (1) man-machine interaction research including large-scale experiments with the success path monitoring system as well as operator training; (2) computer-based operator support systems including the development of systems for early fault detection, disturbance
diagnosis, operator action advice and computerized procedures; and (3) control room design with emphasis on improvements through the use of modern CRT displays and micro-processors.

The nuclear fuel programme began as an effort to obtain data on fuel behaviour under worst-case accident, or design basis accident, conditions. The last of these LOCA fuel tests has been completed in 1986. Emphasis has now shifted to studying fuel behaviour under off-normal transient conditions, and water chemistry problems. As a secondary effort, the associate members study the behaviour of new and advanced fuel designs before they are implemented in large power reactors. The fuel performance experiments and analysis programmes proposed for 1987 comprise four activity areas: (1) base-power and ramp operation examinations including studies on fuel thermal performance, fission gas release properties and pellet-cladding interaction; (2) cyclic and transient operation investigations; (3) extended burn-up operation studies including waterside corrosion of Zircaloy cladding; and (4) fuel performance analysis work.

PRIMARY CIRCUIT INTEGRITY

A central objective of reactor safety is to ensure that no leaks or breaks will develop in the pressure vessel and coolant circuit pipework, which together constitute the primary coolant circuit. If not detected and monitored, defects inherent in these steel components could grow as cracks under the repeated stresses arising from pressure and temperature changes. Such growth can be enhanced by the effect of embrittlement of the steel from the radiation emanating from the reactor core.

In the context of many national and international studies of the issues involved, the NEA programme is focussed on methods, both theoretical and practical, for treating two complementary aspects: fracture mechanics (the assessment of how defects grow), and non-destructive methods for detecting and sizing defects. Increasing attention is also being given to the integrity issues raised by the possible degradation of components with age.

Fracture Mechanics

In fracture mechanics, Member countries have continued to develop methods for calculating how defects develop in steels as a function of the characteristics of the defects themselves, the properties of the steel in which they are located, and the nature of the stresses applied to the system.

These activities provide input to current cooperative studies of several particular issues. These include the influence of residual welding stresses and of stress gradients at surfaces on crack behaviour. A survey will be completed in early 1987 of national approaches taken to the safety assessment of cracks. Vessel and piping failure mode probabilities remain under consideration. Problems in stress analysis have been compiled as the basis of a benchmark exercise to validate stress analysis programmes, to be commenced in 1987. A personal computer based data bank of toughness properties of reactor steels is under discussion. Finally a state-of-the-art-report on stress corrosion cracking will be published in early 1987 and a review report of the effect of weld repairs on integrity is being considered.
Large scale vessel and pipe tests in national programmes will soon produce test results that can be used to validate fracture mechanics analysis methods developed from theoretical principles and laboratory scale tests. A Fracture Assessment Group has been formed to undertake this validation and to develop an analysis benchmark.

Degradation of components with age

As the older LWRs approach the end of their planned service life, increased attention is being given to the factors which might determine a lifetime limit. The capability of pressurised components and their fittings to continue to perform safely as they age is important in this regard. A workshop will be held in early 1987 to identify the technical issues which may limit the long term integrity of such components. It will recommend what further activities on degradation of components should be undertaken within the programme on primary circuit integrity, including a priority topic for a Specialist Meeting in Sweden in late 1987.

Non-Destructive Examination (NDE)

Inspection with ultrasonic waves continues to be the main NDE technique for locating and sizing flaws in reactor primary circuits so that their safety significance may be evaluated. The series of Programmes for the Inspection of Steel Components (PISC) carried out since 1975 under the auspices of OECD-NEA and the Commission of the European Communities is a major effort to better characterise the capability of ultrasonic inspection techniques and procedures. These programmes are centred on the Ispra Joint Research Centre of the Commission of the European Communities who, in their role as Operating Agent, manage the programmes and provide approximately half of the programme funding (the other half coming via contributions in-kind from non-CEC countries). NEA provides the Secretariat of the PISC Managing Board. The second Programme for the Inspection of Steel Components (PISC-II) terminated in 1986 with a PISC Symposium where the results of round robin tests on plates and nozzles by 50 organisations in 14 countries were discussed. The PISC II results increased confidence that techniques exist which have the necessary capability for reliable inspection of reactor pressure vessels. The most capable procedures and techniques were identified and the direction for their optimisation indicated with respect to dealing with particular defect characteristics. One direct consequence of the work will be the adoption in 1987 of important changes in inspection procedures in the rules for inspection of nuclear components of the ASME Pressure Vessel Code.

A number of problems were highlighted by PISC II for the sizing of defects and for the resources needed for effective inspection. Some of the problems and uncertainties are related to human factors. These particular issues are being addressed in a third PISC programme (PISC-III) which commenced in 1985, to amplify, verify and validate aspects revealed in the PISC-II work by considering real defects and real geometries in real surroundings (some radioactively contaminated). Parametric studies and work on modelling of ultrasonic examination commenced in PISC II are being continued. Particular attention is being given to the difficult problems of ultrasonic inspection of stainless steel in round robin tests which will commence in 1987. A screening phase has been completed. The programme also includes round robin inspections of bimetallic welds (nozzle safe ends) and steam generator tubes.
Outside of PISC a report will be completed in early 1987 on the role of round-robin testing in establishing the reliability of ultrasonic examination of thick ferritic steel plate. A state-of-the-art report on steam generator tube plugging criteria is in preparation. One is planned on the monitoring of local loading and service conditions that influence crack behaviour. This will extend the scope of the Report on Continuous Monitoring of Plant Service Conditions published by OECD in 1986.

SOURCE TERM AND ACCIDENT CONSEQUENCES

The dominant issue in nuclear safety over the last few years has been the source term, that is the quantity of radioactive material which might be released in a nuclear reactor accident, its physical and chemical form, and other characteristics needed to completely specify its dispersion in the environment (e.g. height of release, duration of release, energy in the plume, etc.).

Dramatically emphasized by the Chernobyl accident in April 1986, the importance of the source term issue had been accentuated first by the accident that occurred in the Three-Mile Island-2 plant in March 1979. The TMI accident did not produce the fission product leakage considered in the regulatory assumptions made at that time for accident source terms, or expected from the more quantitative source terms of the 1975 U.S. Reactor Safety Study (WASH-1400). The inert gas fission products behaved about as expected, but all other radionuclides were released in surprisingly low amounts, in particular iodine-131. While about six million curies of xenon escaped to the environment, the iodine leakage was only about 18 curies, less than one part in ten million of the iodine in the core. According to classical assumptions, fission product iodine should have been volatile at the high temperatures of a degraded core and should have appeared as a gas, in amounts similar to those of the inert gases.

It was suggested in 1980 that the explanation for the unexpectedly low air-borne release of iodine during the TMI-2 accident was that the reducing conditions in the vessel of a water reactor during a core-damaging accident would transform iodine into non-volatile iodide. Iodine would therefore not appear as a gas. This view triggered a vast programme of experiments and analytical studies on fission product behaviour, for various reactor types and accident sequences. On the basis of these research efforts and analyses it is now believed that for water reactor accidents resulting in core damage, iodine would appear as caesium iodide and the excess caesium (there is about ten times as much caesium as iodine in the fission products) as caesium hydroxide. Both compounds are highly soluble in water and are much less volatile than elemental iodine or caesium. Moreover, they can form aerosols, which are subject to depletion processes in the reactor primary circuit as well as in the containment building. Therefore, iodine and caesium source terms may be much reduced from the WASH-1400 estimates. If this assessment proves correct, it could have major implications, in some cases, for such regulatory issues as plant siting and emergency response planning, since in the event of a major accident, the potential radiation exposure of the neighbouring population would be much lower than previously assumed.

In order to arrive at realistic estimates of accident source terms, and considering the important implications of this issue, CSNI has set up a number of activities in the areas of source term assessment, reactor containment atmosphere control systems, and accident consequence analysis.
The main activity during 1985 had been an intercomparison of the various source term studies published at that time and an analysis of the results of recent work performed in OECD Member countries. The results of the intercomparison were published at the beginning of 1986. Emphasis was placed on areas where considerable progress had been made since WASH-1400, areas where source term information was considered sufficient, barriers to applying source term information to different LWR plants and generic issues applicable to most plants, uncertainties and areas of disagreement between various source term studies, and recommendations for dealing with the uncertainties. The information supporting the conclusions of the intercomparison has been extensively peer-reviewed during 1986 and is now ready for publication.

However, it was recognized that further in-depth information was required in some areas. Some key issues that had not been completely addressed in previous studies were identified, and NEA groups of experts undertook to carry out appraisals to further characterize these issues. These included: fission product chemistry in the primary circuit of a LWR during severe accidents; resuspension/re-entrainment of aerosols in LWRs following a meltdown accident; iodine chemistry under severe accident conditions; effects of combustion, steam explosions and pressurized melt ejection on fission product behaviour; radionuclide removal by pool scrubbing; fission product release from fuel in severe accidents; containment leakage and aerosol behaviour in leak paths.

In general, it is felt that considerable progress has been made, and that source terms can be calculated on a more mechanistic basis. It is also apparent that source terms cover a very wide range of technologies and the various effects cannot be treated in isolation. In particular, there is a continuing need to couple thermalhydraulics and fission product behaviour when assessing the transport of radioactivity during severe accidents. Another important undertaking is to assess the uncertainties associated with the new source term information. A specialist meeting on core debris/concrete interactions was held in 1986; its main purpose was to review and discuss recent progress and to provide a forum for interchange between experts in the fields of heat transfer mechanisms, thermodynamics, and fission product release.

From a more general point of view, current national source term positions and practices in OECD Member countries have been surveyed in recent years; the survey will be updated in 1987. A survey has also been made on the question of filtered vented containment systems (FVCS).

The ultimate aim of safety assessment is to evaluate the radiation dose to the public after radionuclides have been released into the environment. Different ecological exposure pathways are possible; there is the chance, for example, of direct external irradiation from radionuclides in the air, on or in the ground or clothes, of internal exposure after inhalation of contaminated air, or of ingestion of contaminated water or foodstuffs. An evaluation of the main pathway parameters important for accident consequence modelling has been completed and will be further updated; these parameters are: decontamination, radionuclide behaviour in urban areas, shielding factors, filtering effects of houses and deposition indoors, wet and dry deposition, migration of radionuclides in soil, deposition and decontamination under winter conditions, deposition on crops and root uptake. Work is also underway on meteorological sampling techniques, the sensitivity of dose pattern to grid size for population, and the estimation of accident
consequence uncertainties.

TMI-2 EXAMINATION

A Three-Mile Island-2 Examination Programme has been initiated by NEA in collaboration with the U.S. Department of Energy. The purpose of the programme is twofold:

— establish international participation in DOE's TMI-2 Sample Acquisition and Examination Program;

— establish a calculation programme along the lines of an international benchmark exercise, wherein interested countries will analyze various periods of the TMI-2 accident in an effort to assess and improve the relevant severe accident computer codes.

As part of the programme, DOE is making available samples from the TMI-2 core and reactor components from the reactor coolant system and from areas within the reactor building. Of these, the unanimous choice for a first delivery was core bores; core bore samples were selected by an international expert group at the beginning of 1987 and the selected samples will soon be delivered to each participant's hot cells. It is expected that two more shipments of samples will take place at annual intervals.

In connection with the TMI-2 analysis exercise, a package has been provided to all interested countries to help them model the TMI-2 reactor and to initialize their codes to run predictions of the TMI-2 accident. The first workshop to cover modelling and initializing will take place in early 1987.

SEVERE ACCIDENTS

In a reactor accident, the principal concern is that the engineered safety systems will fail, resulting in a large release of radioactive material. The design basis of a nuclear power plant therefore comprises basic specifications which are defined for the designer in order to ensure the capability of the plant to undergo a specified range of operational events, accidents, and external hazards within strictly limited radiological protection requirements. The design basis usually includes the specification of challenging events, important assumptions, and in some cases particular methods of analysis. It is important to remember that a design basis accident is essentially a design tool to help make an engineering judgement on the appropriate safety margins for different component parts and systems of a nuclear plant; strictly speaking, it should never be used to assess accident consequences because of the extreme conservatism placed on the basic assumptions.

From a safety point of view, however, it is necessary to envisage accidents which exceed the design basis sufficiently to cause failure of the structures, materials, systems, etc. without which core cooling cannot be properly assured by normal means. Such accidents are called "severe accidents"; their degree of severity depends on the degree of fuel damage and on the degree of loss of containment integrity. One of the purposes of considering severe accidents is to ensure that sensible account is taken of design features of the plant and emergency procedures in limiting consequences (which could well turn out to be less than those of a design basis accident). With this end in view, "realistic" estimates are needed rather than
conservative estimates, in particular because exaggeration of possible consequences could result in inadequate emergency response and undue safety, health, and economic risks, diversion of time, money and talent from reasonable safety concerns, unjustified expenditure on safety hardware, and erroneous public perception of nuclear power risks.

Member countries with major nuclear power programmes have defined severe accident policies. A Senior Group of Experts on Severe Accidents was set up by the CSNI to study the potential response of existing LWR systems to severe accidents, to examine the implications for current R & D, and to advise the Committee accordingly. A report was published in 1986.

It is generally agreed that current designs of light-water reactors are in fact far more capable of coping with severe accidents than the case of the design basis accident would suggest, and that it is possible to cope with events which go well beyond the design basis provided that appropriate preparations are made to make the most of the resilience of the plant. Consequently, whilst consideration should be given to scope for enhancing existing systems' specifications with a view to effectiveness in accident conditions more severe than the design basis, radical design changes are not thought necessary.

It is also generally agreed that the capability of a plant to function in conditions well beyond the design basis provides a margin of safety which should be exploited to maintain control over events and minimise the consequences to the public. Thus, the most effective approach is to make accident initiation less likely, as well as to reduce the probability of its propagating at every subsequent stage; accident management is of the highest importance at all stages of accident development from initiation to long term control measures.

Highest priority should therefore be given to improving the ability of plant personnel to monitor, diagnose and influence the course of a severe accident from the earliest stages. Operator training programmes must be developed to take into account accident diagnosis and management beyond normal operating transients and incidents; they must be frequently reviewed, to assimilate the most up-to-date knowledge. This presupposes an adequate understanding of the physical processes taking place.

As a rule, most Member countries advocate the use of any available means on the plant to control severe accidents. At the same time, it is to be noted that, in spite of plant-specific features and different national practices which lead in particular to the adoption of individual supplementary measures, approaches to seeking ways of intervening at all stages of accidents with a view to establishing control over them are on the whole of a similar nature.

CONTAINMENT

The role of reactor containments as a vital barrier to the dispersion of fission products in the environment has been recognised for some time. Moreover, source term experts have stressed in recent years the importance of the time and mode of containment failure for the estimation of source terms and accident consequences. Although there is accumulating evidence that containments are stronger than thought in the 1975 U.S. Reactor Safety Study (WASH-1400) and that they can withstand pressures which are significantly
higher than their design values, important generic topics justify increased international exchanges in the following areas:

- containment design;
- containment loading and capabilities beyond design;
- containment response;
- containment performance aspects of accident management.

Recommendations for international collaboration in these areas have been made by a group of experts; several activities are underway or in preparation.

RISK ASSESSMENT

Probabilistic Safety Assessment (PSA) is being applied to a steadily increasing range of technical questions in the design and regulation of nuclear power plants. Several comprehensive PSAs have been completed on several reactor types in different countries in recent years. Experience is showing that even partial risk and reliability assessments can provide useful insights into particular safety questions. Analytical methods are being developed to allow for the inevitable uncertainties in risk assessments when the results from them are used in making safety-related decisions.

NEA reviews were completed and published in 1986 on: a survey of the analytical techniques that have been used in PSAs; and on how the results from a variety of risk and reliability assessments have been applied. The studies identified areas where the methods need improvement and also provided insights on how, and where, the conclusions from PSAs at different depths can be used with confidence.

During 1986 a new project started which will provide a state of the art report, by the end of 1987, on the use of reliability engineering techniques in the safety management of nuclear power plants. This task was initiated at a workshop held in the UK at which representatives of some nuclear utilities involved in this field met, along with representatives from the nuclear regulatory bodies in Member countries. National efforts towards developing explicit safety goals for nuclear power production are being continually monitored at meetings of the working group on risk assessment, along with an evaluation of the technical problems that are involved in attempting to define such safety standards. Work has also continued on the difficult problems associated with quantifying and including the human effect in risk assessments, and on methods for evaluating the importance of human interventions on reactor behaviour during accidents.

CHERNOBYL

Much time and effort was devoted in recent months to the study of the Chernobyl accident and its consequences. Although it became quickly clear that Chernobyl-type reactors were very different from commercial reactors in operation in OECD Member countries, the accident has had some radiological impact and quite a societal impact in OECD countries. It was therefore very important to understand, as fully as possible, the causes of the accident, the characteristics of the radioactive release and of its dispersion. It was also
important to analyse any possible lessons which might be drawn for the safety of reactors in OECD countries, and for the implementation of emergency plans should a major nuclear accident happen.

The main conclusion of the work undertaken under the auspices of CSNI is that no new information or phenomena have been identified which would require modification to reactors in OECD countries. Indeed, the principal lessons which can be drawn from the Chernobyl accident have already been learned by OECD countries several years ago, as a result of the accident at the Three Mile Island -2 plant, and translated into safety improvements. Nevertheless, Chernobyl will lead to increased cooperation among OECD countries in the area of human factors and in detailed analysis of reactor incidents.

FUEL CYCLE SAFETY

Although the nuclear reactor receives the bulk of safety attention, consideration must also be given to the industrial facilities that are required to produce, recycle and dispose of the nuclear fuel elements themselves.

In recent years NEA has conducted benchmark exercises on the computer programs that are used to make criticality calculations for spent reactor fuel in shielded transport casks, and for stacks of packages containing the various fissile materials that are used in fuel fabrication and reprocessing.

A CSNI Working Group reviews every two years the state-of-the-art concerning the safety principles, practices and technologies which are applied to the whole range of fuel cycle facilities. The latest review was carried out in late 1985 and the next is planned for 1987. Specialist Meetings in this area are also held on topical subjects and in 1986 one such meeting was held in Salamanca, Spain, to systematically compare the regulations concerning the treatment of external events such as earthquakes, fires and missile impacts, which threaten the safe operation of nuclear fuel cycle facilities in Member countries.
Annex I: NEA Staff Dealing with Nuclear Safety Matters (Spring 1987)

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Reactor Transients and LOCAs: Ralph Caruso (Tel. 45 24 96 82)
TMI-2 Examination Activities: Jacques Royen (Tel. 45 24 96 92)
OECD LGFT Project: Ralph Caruso (Tel. 45 24 96 82)
OECD Halden Project: Ralph Caruso (Tel. 45 24 96 82)
Primary Circuit Integrity: Neil McDonald (Tel. 45 24 96 79)
OECD/CEC PISC Programme: Neil McDonald (Tel. 45 24 96 79)
Source Term and Accident Consequences: Jacques Royen (Tel. 45 24 96 92)
Containment Performance: Jacques Royen (Tel. 45 24 96 92)
Risk Assessment: John Caisley (Tel. 45 24 96 74)
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Annex II: Recent NEA Publications in the Nuclear Safety Area

- Air Cleaning in Accident Situations (Report by a Group of Experts); General Distribution
- Nuclear Aerosols in Reactor Safety (Report by a Group of Experts); General Distribution
- Nuclear Aerosols in Reactor Safety (Proceedings of a Specialist Meeting); Restricted Distribution
- International Comparison Study on Reactor Accident Consequence Modelling (Report by a Group of Experts); General Distribution
- Nuclear Reactor Accident Source Terms (Report by a Group of Experts); General Distribution
- Technical Annexes to Nuclear Reactor Accident Source Terms; Restricted Distribution
- LWR Containment Aerosol Code Comparison Exercise (Report by a Group of Experts); Restricted Distribution
- Iodine Chemistry in Reactor Safety (Proceedings of a Specialist Workshop); General Distribution
- Severe Accidents in Nuclear Power Plants (Report by a Group of Experts); General Distribution
- Steam Generators (Proceedings of a NEA/UNIPEDE Specialist Meeting); Restricted Distribution
- On-Site Electric Power Sources (Proceedings of a Specialist Meeting); Restricted Distribution (in preparation)
- Interaction of Fire and Explosion with Ventilation Systems in Nuclear Power Plants (Proceedings of the Second Specialist Meeting); Restricted Distribution
- Probabilistic Methods in Seismic Risk Assessment for Nuclear Power Plants (Proceedings of the Second Specialist Meeting); Restricted Distribution
- Standard Problem Exercise on Criticality Codes for Large Arrays of Packages of Fissile Materials (Report by a Working Group); Restricted Distribution
- Defect Detection and Sizing (Proceedings of a NEA/IAEA Specialist Meeting); General Distribution
- Leak-Before-Break (Proceedings of a Specialist Meeting); Restricted Distribution
- Fourth Symposium on Reactor Noise (SMORN-IV Proceedings); General Distribution
- Ductile Piping Fracture Mechanics (Proceedings of a NEA/USNRC Workshop); Restricted Distribution
- Decay Heat Removal Systems (Proceedings of a Specialist Meeting); General Distribution
- Water Reactor Containment Safety (Proceedings of a Specialist Meeting); General Distribution
- Activities on Human Factors (Newsletter no. 2); General Distribution (in preparation)
- Design and Implementation of Emergency Operating Procedures (Report by a Group of Experts); Restricted Distribution
- Expert Judgement of Human Reliability (Report by a Group of Experts); Restricted Distribution
- Nuclear Safety Research Index 1984; Restricted Distribution
- Catalogue of Test Specimens for NDE; General Distribution
- The Ultrasonic Inspection of Austenitic Materials; General Distribution
- Specifications for the Digital Recording of Ultrasonic Signals; General Distribution
- Near Surface NDE of Reactor Steels; General Distribution
- Stable Crack Growth during Overstressing in Proof Testing of Pressure Vessels; General Distribution
- Design of Pressure Suppression System Containments (Report by a Group of Experts); General Distribution
- Core Debris/Concrete Interactions (Proceedings of a Specialist Meeting); Restricted Distribution
- Chernobyl and the Safety of Nuclear Reactors in OECD Member countries (Report by a Group of Experts); General Distribution (in preparation)
- Continuous Surveillance of Reactor Coolant Circuit Integrity (Proceedings of a Specialist Meeting); General Distribution
- A Summary of the PISC-II Project, PISC-II Report No. 1 Final/Issue (CEC-JRC Ispra/OECD-NEA); Restricted Distribution
- Source Term Assessment, Containment Atmosphere Control Systems, and Accident Consequences (Report by a Group of Experts); Restricted Distribution
- Selected Source Term Topics (Report by a Group of Experts); Restricted Distribution
Annex III: Outline of NEA's Main Activities

The work programme of the Agency is divided between two main sectors. One is concerned with questions regarding the safety and regulation of nuclear activities, for which the Radiation Protection and Waste Management Division, the Nuclear Safety Division and the Legal Affairs Section have been established. The other covers the scientific and economic aspects of nuclear energy, for which the Nuclear Development Division, the Nuclear Science Division, and the NEA Data Bank have been established.

RADIATION PROTECTION

NEA's primary purpose in this field is to contribute to a better understanding of the phenomena and problems involved in the exposure of man to radiation and of the potential health effects resulting from nuclear energy activities.

Specific areas of activity include: occupational exposure of workers in the nuclear industry; control of radioactive effluent discharges from nuclear installations into the environment; protection of uranium miners and consideration of environmental problems associated with uranium mining and milling; emergency planning for nuclear facilities, etc.

RADIOACTIVE WASTE MANAGEMENT

NEA has developed a comprehensive programme on radioactive waste management, a field concerned with the handling, treatment and disposal of radioactive wastes generated by the nuclear fuel cycle.

The area of greatest concern is the disposal of high-level and long-lived radioactive wastes, and the option receiving most attention is the emplacement of such wastes in deep geologic formations either on land or on, or beneath, the ocean floor. The NEA programme includes study of appropriate disposal methodologies and analysis of the long-term risks associated with geologic repositories. Particular attention is given to factors important in site-specific assessments. Having regard to the long-term considerations involved, questions concerning the administration and surveillance of emplacement sites are also being discussed.

The Agency has also responded to problems which have grown in importance as the industry has matured and the volumes of waste have increased. These include the treatment and disposal of intermediate level, airborne and combustible solid wastes, the long-term management of uranium mill tailings and the decommissioning of nuclear facilities at the end of their useful lives.

Under a special Mechanism established by the OECD Council in 1977, the Agency provides for international consultation and surveillance of the disposal by Member countries of packaged low-level wastes into the deep ocean.

NUCLEAR SAFETY

The Nuclear Safety Division's primary task is to arrange international co-operation in research relating to the safe operation of nuclear power reactors and fuel cycle facilities and in associated regulatory questions. More details are given in the main part of this report.
NUCLEAR LAW

One of NEA's traditional objects has been to foster international co-operation in nuclear law with a view to promoting harmonisation of legislation governing nuclear activities.

NUCLEAR FUEL CYCLE STUDIES

NEA activities in this field are conducted in close collaboration with the International Atomic Energy Agency (IAEA) in Vienna and OECD's International Energy Agency (IEA). These are concerned primarily with assessment of the technical and economic characteristics of nuclear energy growth and the nuclear fuel cycle, in particular:

- assessment of uranium resources, production and demand;
- nuclear power growth projections and associated nuclear fuel cycle requirements and supply consideration;
- economics of the nuclear fuel cycle;
- technical studies on nuclear energy development.

NUCLEAR SCIENCE

NEA provides a framework for co-operation and exchange of information on basic nuclear data and provides supporting services to assist Member countries on matters relating to reactor physics and nuclear data.

NEA DATA BANK

The NEA Data Bank allows members to share the large and complex computer programs used in the design and operation of reactors.

The Data Bank also belongs to a world-wide network of four data centres which exchange neutron cross-section data and present bibliographic references and the data themselves in a form convenient for research laboratories and other users.
ANNEX II

Information on
Advanced Containment Experiments (ACE)
Programme
ADVANCED CONTAINMENT EXPERIMENTS (ACE) PROGRAM

PROGRAM OBJECTIVE

A new Advanced Containment Experiments (ACE) Program is proposed to be performed at Argonne National Laboratory and in the Containment Systems Test Facility (CSTF) at Battelle-PNNL, Hanford Engineering Development Laboratory, with technical support and related experiments provided by supporting organizations. The ACE program will complement the current LWR Aerosol Containment Experiments (LACE) Program by investigating additional fission product production and deposition phenomena involved in degraded reactor containment building (RCB) conditions and means of controlling fission product release using passive filtering devices. The objectives of the ACE program are: (1) to provide a comparative experimental basis for filtration techniques (e.g., submerged gravel beds, water pools, sand beds, etc.), (2) to provide data for modeling radioactive iodine species transport, (3) to determine fission product releases from molten corium concrete reactions, and (4) to develop and validate the applicable computer codes and models.

INTRODUCTION

Extensive research is presently underway around the world to understand the behavior of radioactive materials in both the primary system and the containment building of an LWR under postulated severe accident conditions. The objective of this research is to characterize aerosol production and attenuation processes which, in the event of a postulated severe accident, effect the radioactive source term.

Early experiments (Containment Systems Experiment) provided valuable information on fission product transport and retention by natural processes and the effect of engineered safety features on fission product removal. Three aspects, however, were not realistically accounted for in the CSE tests. These were: (1) very high aerosol concentrations, (2) transient thermal conditions, and (3) the effect of intercompartmental flows.

The recent Marviken Program was directed towards demonstrating radioactive aerosol retention in reactor coolant systems while the DEMONA Project seeks to demonstrate inherent aerosol attenuation in intact single and multicontainments and to demonstrate the predictive capabilities of the NAUA aerosol code and COCMEL thermal-hydraulic code. The DEMONA experiments emphasized intact containment buildings and relatively steady-state thermal hydraulic conditions. Pilot-scale, single-compartment tests have also been performed in the ORNL Nuclear Safety Pilot Plant (NSPP) and in the NAUA facility at KFK to study aerosol behavior for code development and validation.

In the area of molten corium/concrete interaction (MCCI) studies, two large, related programs exist. The first, the BETA program performed in the FRG, involved dropping up to 450 kg of Fe-Al$_2$O$_3$ melt into concrete crucibles. Internal heat generation was sustained by inductive heating of metallic constituents of the melt. The recently completed program consisted of nineteen tests; sixteen using basaltic concrete, three using limestone concrete and focused on the thermal hydraulic aspects of the interaction including 2-D concrete erosion phenomena.
The second program is ongoing at Sandia National Laboratory and consists of five related series of experiments:

- **TURC** - Transient drop of molten stainless steel or corium onto concrete; no sustained heat generation.
- **SWISS** - Steady-state induction heating of stainless steel atop concrete.
- **HS** - Solid phase penetration of a heated block into concrete.
- **WITCH/ GHOST** - Steady-state induction heating of steel with gas sparging to study mechanical/condensation aerosol formation processes.
- **SURC** - Steady-state induction heating of stainless steel or corium (IRIS heating atop concrete. Four 250 kg corium tests are planned using limestone/limestone and baltic concrete, with and without water addition atop melt. The SURC large-scale corium tests have not begun although small-scale, developmental tests are under way.

The proposed ACE experiments will complement the BETA and SNL programs — extending the range of materials studied and focusing on the fission product releases from the corium concrete interaction.

Currently, the LWR Aerosol Containment Experiments (LACE) Program is investigating inherent radioactive aerosol behavior for postulated high-consequence accident situations. The LACE program extends the Marviken database to containment building phenomena, and complements the DEMONA tests by considering RCBs that are no longer intact. Accident situations being considered in LACE are those for which high consequences are presently calculated because either the containment is bypassed altogether, the containment is impaired early in the accident (leakage, failure to isolate), or containment degradation occurs simultaneously with a large fission product release. The significant inherent aerosol retention being observed reduced the consequences presently calculated for these postulated accident situations. The results of the LACE program considerably improve our ability to realistically assess the consequences of presently assumed high consequence accidents.

The LACE program is sponsored by an international consortium, organized by EPRI, consisting of fifteen sponsors. The lead organizations of the LACE consortium are: Commission of the European Communities; OH (Canada); VTT (Finland); CEA (France); BMFT (Federal Republic of Germany); ENEA (Italy); JAERI (Japan); KEMA (The Netherlands); UNESA (Spain); SKI (Sweden); EIR (Switzerland); AEA (United Kingdom); and EPRI, DOE, NYPA, and NRC (United States).

The LACE program consists of three main tasks.

- **Large-Scale Tests.** These tests experimentally investigated inherent aerosol retention behavior for conditions which simulated selected high-consequence accident conditions. They also provided a database for validating containment aerosol and related thermal-hydraulic computer codes.
Computer Code Validation. Both pretest and blind posttest calculations using aerosol behavior and thermal-hydraulic computer codes are being compared to experimental data.

Support Programs. The support programs provide direct support needed to perform or interpret the large-scale experiments. The support program also includes separate effects tests needed to clarify individual phenomena.

The computer code validation efforts and support programs are being conducted by many groups within the sponsoring organizations.

As a result of the LACE program, several of the sponsoring organizations have suggested proposals to extend the area of investigation. In particular, additional aspects of fission product retention in RBs and means of passive filtering were thought important. At the LACE Project Board meeting of October 1, 1986, the Electric Power Research Institute (EPRI) was asked to take the lead in putting together this proposal, with input from the LACE cosponsors, to see if a sound technical basis could be proposed and a schedule and budget developed. Consequently, EPRI has determined that significant interest exists in several areas where large-scale containment aerosol experiments would improve our ability to assess the production of and subsequent behavior of fission products (particularly iodine) in containments and to filter fission products prior to release to the environment. In addition, priorities were assigned concerning important physical phenomena that should be studied and test measurements that should be made in order to maximize the benefits of the ACE program. A principal conclusion of our discussions is that the proposed experimental program could be significantly improved by an international approach like that used in LACE that would allow the analytical and experimental expertise of several international organizations to focus on the work. Properly organized, the proposed program could thereby address many of the research requirements necessitated by new safety initiatives such as those being pursued in Europe concerning containment filtering and hydrogen control.

The discussions focused on those cosponsor proposals where significant gains could be made by understanding inherent radioactive aerosol retention processes and which could be viewed as a logical extension of the current LACE program. Three research areas were identified as most important. They are:

1. Containment Filtration. There appears to be a great deal of interest concerning the submerged gravel bed filter and related sand bed and pool scrubbers for fission product removal during accident depressurization situations. Several new concepts exist including submerged Venturi scrubbers developed by ASEA-ATOM, a wire mesh filter developed by KWH and a small-size, submerged gravel bed filter (SGS) concept originally developed by the U.S. DOE for sodium aerosols. The proposal is to develop a database for comparing the merits of pool, sand/ gravel bed, Venturi and SGS filters. About one quarter the proposed ACE program would be related to parametric study of these concepts.
2. **Integral Iodine and Fission Product Retention Tests.** Integral tests have been proposed to demonstrate retention of iodine and fission products in the primary system, containment system, and simulated release to the environment. Emphasis will be on several iodine species and their most important reaction partners, such as cesium, boron, and silver. These tests will provide data regarding iodine species transfer and retention not obtained in either the Marviken, DEMONA, or LACE programs on aerosol behavior. Aspects of hydrogen management techniques (igniters and recombiners) may be addressed if resources exist. This effort would constitute about one quarter of the ACE program.

3. **Large-Scale Reactor Material and Fission Product Release MCCI Experiments.** It has been proposed to investigate the thermal-hydraulic, chemical, and aerosol release processes of molten-core-concrete interaction (MCCI) in a series of large-scale, reactor-material tests and related analyses. The MCCI investigation is justified because of the large contribution that MCCI is calculated to make to the magnitude of and uncertainty in the overall source term from postulated severe accidents. The recently issued report NUREG 1150, identified “The magnitude of the core-concrete release was a key source of uncertainty for all plants.” This effort would be about half of the program.

The general consensus was that if the proposed experiments are of interest, a separate project would be started following the current LACE project. The initial estimate is that if the above work is undertaken, it would require somewhat more time and funding than the current LACE project, namely, about thirty to thirty-six months and $6.5 million, plus the usual support effort. If only part of these proposals were implemented, the time and resources would be scaled back accordingly. Preliminary ideas on the program had also been discussed at the last LACE Project Board and Technical Advisory Committee meetings.

Based primarily on these discussions, EPRI has prepared this Program Plan. It is intended to provide a proposed test program. As is the custom in the LACE program, the ACE series of tests could be redirected, if necessary, by a decision of the Project Board, and considerable latitude would be given to a Technical Advisory Committee to set the technical aspects of the experiments.

**PROPOSED ACE PROGRAM**

The experimental test matrix proposed below for the ACE program includes a base program and other support elements necessary to establish a computer code and model development and validation program. The ACE program has been divided into three phases to facilitate its implementation and to encourage international participation. The program organization will maximize use of international facilities and should minimize out-of-country costs to the participants. The large-scale experiments at ANL and HEDL will form the base experimental program. The base program will be complemented by smaller-scale
experiments, calculations, and analytical efforts at other laboratories (called the Support Program). The Support Program will include tasks in chemistry, instrumentation, thermal hydraulics, and aerosol mechanics that are vital to the success of the overall program.

The management of the ACE program will be vested in a Project Board of Directors (PB), comprised of representatives from each program participant. Voting on the Project Board will be proportional to each participant's share of the overall program. A Technical Advisory Committee (TAC) will be established to monitor the technical progress of the project, to advise the PB on the program direction, and to coordinate the base and support program efforts. The PB and TAC are expected to meet about twice a year.

**CSTF DESCRIPTION**

The aerosol attenuation tests of the ACE program would be performed in the Containment Systems Test Facility (CSTF). This facility was formerly used for the Containment Systems Experiment (CSE) and is the same one used for the LACE program. The CSTF consists of a large laboratory area (over 1400 m² of floor space) that houses a large model containment vessel. The major features of the CSTF are listed in table 1.

<table>
<thead>
<tr>
<th>Table 1</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>CONTAINMENT SYSTEMS TEST FACILITY (CSTF) FEATURES</strong></td>
<td></td>
</tr>
<tr>
<td>Total Height (m)</td>
<td>20.3</td>
</tr>
<tr>
<td>Diameter (m)</td>
<td>7.6</td>
</tr>
<tr>
<td>Volume (m³)</td>
<td>852</td>
</tr>
<tr>
<td>Design pressure (MPa)</td>
<td>0.52 at 160°C</td>
</tr>
</tbody>
</table>

In addition to a large equipment access hatch and a smaller personnel access hatch, numerous flanges exist on the vessel for electrical and mechanical equipment penetrations. In-vessel and ex-vessel sampling is possible for gas analysis, aerosol characterization, and containment thermal-hydraulic characterization. The laboratory also contains various equipment, including data acquisition systems. In addition, Westinghouse Hanford Company (WHC) has an experienced staff who has been involved for many years in aerosol behavior and thermal-hydraulic studies.

The CSTF has been used for a breeder reactor Aerosol Behavior Code Validation and Evaluation (ABCOVE) Program support by the U.S. Department of Energy (DOE) as part of the Fast Reactor Safety Program. In addition to the ABCOVE program, WHC has developed the submerged gravel scrubber under DOE funding for use in filtering sodium aerosols in fast reactor accident scenarios.
Experimental Measurements and Techniques

The measuring equipment and experimental techniques to be used in the ACE program are essentially identical to those developed and used in the LACE program. The aerosol generation techniques will be the same, except some additional work will be required for generation and measurement of the iodine species. However, the project has had considerable experience in these techniques due to previous work on CSE and the fast reactor experiments so that no special difficulty is envisioned. However, iodine sampling and chemical analysis methods will need to be developed.

Filtration Equipment Description

Over the past fifteen years, HEDL has tested many filtration concepts including sand/gravel bed, Venturi and submerged gravel scrubber filters. Much of the equipment needed to do a comparative testing program exists on-site, and many of preliminary scoping experiments needed for ACE have already been performed as part of the other program. A quench tank, Venturi and fibrous scrubbers are already in place as part of the LACE facility and have been used in prior research programs studying iodine species (figure 1). In particular, several special and unique devices exist related to the submerged gravel scrubber concept. These are:

<table>
<thead>
<tr>
<th>Scale</th>
<th>Capacity (m³/sec)</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Bench-scale size units</td>
<td>Small</td>
<td>-0.03</td>
</tr>
<tr>
<td>2. CSTF-SGS</td>
<td>Intermediate</td>
<td>0.5</td>
</tr>
<tr>
<td>3. Large Sodium Fire SGS Facility</td>
<td>Full</td>
<td>4.7</td>
</tr>
</tbody>
</table>

Modification of the existing facilities can be accomplished relatively easily to do parametric testing called for in this proposal.

The project plans, as recommended by the TAC, to consult with various filtration system vendors to acquire and construct other test devices. TAC specifically recommended interactions with Westinghouse, ASEA-ATOM and KWU. The goal of these interactions is not to test proprietary hardware or systems, rather to use commercial input in guiding the test matrix so that key phenomena which form the technical basis of commercial systems can be confirmed and evaluated.

Testing under realistic LWR conditions is a major gap in the existing database for all these devices. The extent of vendor participation in the program is
not settled at this point since access to the data and the distribution of results to the vendors depends on the participation of various co-sponsoring organizations, if the normal project rules concerning the proprietary ACE information are met.

Phase A - Comparative Filtration Tests

The objective of phase A is to obtain decontamination factors for five fission product filtration system currently available: pool scrubbers, standard SGS systems (with demister), combined pool/SGS systems, Venturi scrubber backed up with a pool, fibrous metallic filter, and sand/gravel beds. The proposed experimental details are expanded to obtain a self-consistent comparison of the strengths and weaknesses of each system. A preliminary (mostly qualitative) evaluation of the strengths of the concepts is given in Table 2. These conclusions, however, need to be verified and quantified under realistic LWR accident conditions.

<table>
<thead>
<tr>
<th>Type</th>
<th>Mass Loading</th>
<th>Efficiency</th>
<th>Passivity</th>
<th>Decay Heat Capacity</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Dry sand/gravel bed</td>
<td>Low</td>
<td>High</td>
<td>Yes</td>
<td>Low</td>
</tr>
<tr>
<td>2. (Deep) pool scrubber</td>
<td>High</td>
<td>High</td>
<td>Yes</td>
<td>High</td>
</tr>
<tr>
<td>3. SGS (with Demister)</td>
<td>High</td>
<td>Very high</td>
<td>Yes</td>
<td>Low</td>
</tr>
<tr>
<td>4. Combined Pool/SGS</td>
<td>High</td>
<td>Very High</td>
<td>Yes</td>
<td>High</td>
</tr>
<tr>
<td>5. Combined Venturi/Pooll</td>
<td>High</td>
<td>High</td>
<td>Yes</td>
<td>High</td>
</tr>
<tr>
<td>6. Fibrous Metallic Filter</td>
<td>Low</td>
<td>High</td>
<td>Yes</td>
<td>Low</td>
</tr>
</tbody>
</table>

*Inclusion of a filter system in the final test matrix depends on cosponsorship of the ACE program and resulting access to vendor information.

BASE PROGRAM TEST MATRIX

The rated gas flow rate of each system to be tested will be approximately 0.1 m³/sec. Steam injection to the aerosol mixing vessel will be used to heat any water pools to simulate effects of decay heat. Thermocouples will be provided to measure the water pool temperature where appropriate.
Aerosol characterization stations will be provided at the inlet and outlet to the test devices. The total efficiency or decontamination factor of the system will be measured. This total efficiency will include the efficiency of the device. When water pools are used in combination with other systems, the total efficiency includes the efficiency of water pool scrubbing less the decrease in efficiency which may result from entrainment of aerosol material at the pool surface.

Aerosol characterization at the inlet and outlet of the system will include measurement of the material concentration by filter samples and aerosol particle size by cascade impactors as functions of time. After each test run, the entire system will be cleaned and a material mass balance will be completed, including the integral mass collected on a downstream absolute filters.

Aerosol collection efficiencies will be measured by two methods. First, collection efficiencies as a function of time will be determined by comparing aerosol inlet concentration, aerosol outlet concentration, and soluble aerosol material collected in the water pool during each time step. Second, the integrated collection efficiency over the entire test run will be measured by completion of the mass balance. The integrated collection efficiency will be compared to the average collection efficiency for each time step.

Aerosols to be Used

Three types of aerosols will be used initially. For each test run, the aerosol will be generated in an aerosol mixing vessel (AMV) and then directed through the test unit at the desired gas flow rate.

Cesium hydroxide (CsOH) will be generated by a vapor condensation method. Cesium metal will be vaporized in a furnace and will be injected into the AMV that will contain steam and nitrogen. The cesium vapor will react with steam to form CsOH and hydrogen. The hydrogen concentration will be monitored in the AMV to provide an on-line indication of the variation in CsOH concentration. Based on previous experience, the CsOH aerosol will have an aerodynamic mass median diameter (AMMM) of about 1.8 microns and a geometric standard deviation of about 1.8. Both the AMMD and standard deviation will be measured during each test by cascade impactors.

Cesium iodide aerosol will be generated by vaporizing CsI in a furnace. The CsI vapor will be injected into the AMV where, in the presence of nitrogen and steam, the CsI will nucleate into small particles. Based on previous experience with NaI, it is anticipated that the CsI AMMD will be approximately 1.0 microns with a standard deviation of about 1.6. Both the AMMD and standard deviation will be measured during each test by cascade impactors.

Elemental iodine will be generated by vaporizing elemental iodine in nitrogen and injecting it into the AMV. The steam/nitrogen/iodine mixture will then be directed through the test unit at the desired gas flow rate.

Manganese oxide (MnO) aerosol will be generated by vaporizing MnO powder in plasma torches. The METCO 80 kW plasma units developed in the LACE experiments will be used. Temperatures well exceeding 3000°C cause the MnO to vaporize. When it reaches the cooler regions of the AMV, condensation leads
to aerosols with an AMMD of about 1.5 microns and standard deviation of about 1.8. The optimization work done under the LACE program indicates that relatively steady aerosol production can be achieved.

The proposed tests for phase A include ten experiments. All will use identical thermohydraulic conditions resulting from a steam/nitrogen carrier gas injected during a 60-minute period. The data will be used to compare concepts and bridge existing databases for each concept. The phase A tests will fill in the test matrix presented in Table 3. Each of these tests will use CsOH (AMMD = 1.8 µm; σ = 1.8), CsI (AMMD = 1.0 µm; σ = 1.6), and MnO (AMMD = 1.5 µm; σ = 1.8). The experiments will be done under ambient or saturated water conditions in the system under study.

Table 3

<table>
<thead>
<tr>
<th>Concept</th>
<th>Conditions</th>
<th>Ambient Water</th>
<th>Severe Case Water</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pool Scrubbing</td>
<td>5-m deep pool</td>
<td>A1</td>
<td>A2</td>
</tr>
<tr>
<td>Standard SGS</td>
<td>0.6-m deep bed</td>
<td>A3</td>
<td>A4</td>
</tr>
<tr>
<td>(w/demisters)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Combined Pool/SGS</td>
<td>5-m deep pool</td>
<td>A5</td>
<td>A6</td>
</tr>
<tr>
<td></td>
<td>1-deep bed</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sand/Gravel Bed</td>
<td>5-m deep bed, appropriate cross section</td>
<td>A7</td>
<td>A8</td>
</tr>
<tr>
<td>Combined Venturi/ Pool</td>
<td>5-m deep pool, active Venturi device</td>
<td>A9</td>
<td>A10</td>
</tr>
<tr>
<td>Fibrous Metal</td>
<td>5-m long element, appropriate cross section</td>
<td>A11</td>
<td>A12</td>
</tr>
</tbody>
</table>

The objective of the initial phase A tests is to determine the efficiency versus particle size at ambient and saturated pool conditions for five filtration concepts. Other parameters also need to be studied to extend the data for use in practical situations. Separate effects tests are proposed in the phase A continuation that would examine the following issues: decay heat, level of water pools, noncondensible gas to steam ratio, removal efficiency for volatile species (I₂, CH₃I, etc.), hydrogen burning, effect of pH, and long-term volatilization of iodine in sand (silica) beds. Such tests would be intended to validate the performance of design concepts and to help evaluate commercial filtration units. Because phase B is intended to be more design...
specific than phase A, its implementation would depend on working with vendors of filtration and scrubbing units to define a final testing program. It is therefore not possible to present a final test program at this time since more discussions are required. A preliminary test program, however, is given in table 4. It emphasizes volatile iodine, hot temperatures, high noncondensible gas ratios and low/intermediate pool depths. Such conditions are the ones likely to be the most severe test of a filtration system under accident conditions. It is believed that a further refinement of the test matrix will be made in connection with discussions with filtration equipment vendors prior to the experiments. In addition, small-scale experiments may need to supplement the program in table 4 to examine long-term revolatilization issues, coupled with an evaluation of the sensitivity of each system to hydrogen burning issues.

Tables 3 and 4 are viewed as a starting test matrix for phase A of the ACE program, the final test matrix to be defined by the TAC.

**Phase B - Iodine Retention Experiments (IREX)**

**Purpose of Experiment.** Demonstration of retention of fission products, in particular, iodine and cesium in the primary systems, containment and/or auxiliary buildings during a severe core melt accident.

Both laboratory experiments and theoretical modeling indicate that, due to the chemical processes involved in fission product release and transport during a severe accident, little or no release of elemental iodine to the environment is to be expected. Large-scale experiments on an involuntary basis, as severe accidents might be called in this context and which occurred up to date, seem to support this conclusion. Since iodine (and its precursors) may contribute significantly to the radiation risk during a severe accident, the proposed experiment would concentrate on this element and its most important reaction partners, such as cesium, boron, and perhaps silver. Furthermore, because both silver and other metals such as iron will interact with vapor and aerosol species of iodine, the simulation of prototypic aerosols and small-scale support experiments may have to be part of the experiment. Theoretical modeling of iodine behavior under severe accident conditions usually does not take into account iodine plate out on surfaces, which might be a significant retention mechanism and should be quantified by the proposed experiment.

**Sources and Sinks of Iodine During a Severe Accident.**

**Sources.** The single most important source of iodine is the fuel elements as they are heated up and subsequently melted during a core melt accident. The time-dependent iodine release can be calculated under the assumed temperature history of the core heat-up phase and using measured release constants. However, because of its volatile nature a short release phase, during which all of the iodine is released into the primary system, will be a realistic assumption.

**Sinks.** There are several important sinks of iodine in a reactor. First, the important reaction products have to be considered. The fission product cesium is released from the fuel with a tenfold stoichiometric surplus. Cesium will form CsOH and CsI, the latter being deposited in sump water and dissociate there to Cs + and I- ions. Under certain
<table>
<thead>
<tr>
<th>Test</th>
<th>Filtration Type</th>
<th>Pool Depth</th>
<th>Aerosol</th>
<th>Temperature</th>
<th>Noncondensible Gas Ratio</th>
</tr>
</thead>
<tbody>
<tr>
<td>A13</td>
<td>Combined Pool/SGS</td>
<td>low</td>
<td>volatile Iodine(s)</td>
<td>saturated</td>
<td>90%</td>
</tr>
<tr>
<td>A14</td>
<td>SGS with demister</td>
<td>low, high pH</td>
<td>volatile Iodine(s)</td>
<td>saturated</td>
<td>90%</td>
</tr>
<tr>
<td>A15</td>
<td>Combined Venturi/Pool</td>
<td>low</td>
<td>volatile Iodine(s)</td>
<td>saturated</td>
<td>90%</td>
</tr>
<tr>
<td>A16</td>
<td>Sand/gravel Bed</td>
<td>--</td>
<td>volatile Iodine(s)</td>
<td>high</td>
<td>90%</td>
</tr>
<tr>
<td>A17</td>
<td>Fibrous Filter Element</td>
<td>--</td>
<td>CsOH + CsI + MnO</td>
<td>high</td>
<td>90%</td>
</tr>
<tr>
<td>A18</td>
<td>Fibrous Filter Element</td>
<td>--</td>
<td>volatile Iodine(s)</td>
<td>high</td>
<td>50%</td>
</tr>
</tbody>
</table>
conditions, cesium released during a core melt scenario may encounter boron leading to the formation of cesium and boron compounds, further complicating the iodine chemistry.

Silver, present in PWRs mainly as a control rod material, will most likely melt and flow down into the lower core support structure region where it may be evaporated after being reheated by slumping core material. Therefore, it will be released into the containment building after cesium and iodine have been released, and will only then be available to react with iodine to form AgI. AgI is not soluble in water and therefore provides a very stable sink for iodine.

Elemental iodine will partition between the water and gas phase according to relatively well-known laws, which have been verified both in clean condition laboratory experiments and from first principle thermodynamic calculations.

Elemental iodine will also deposit on structural surfaces in the containment and may be retained there, though no quantitative statement can be made at this time about the efficiency and reversibility of this effect both as a function of surface type and thermodynamic parameters.

A very large surface is also available for deposition as long as aerosols are present during an accident. In addition, iodine will absorb into paint.

Proposed Experiments. The experiments proposed in Phase D are intended to link directly to the experiments in the LACE program. Three tests are proposed. The first and second tests are essentially repeats of test LA-1 using CsI as the dominant aerosol species (refer to document LACE TP-1 for a description of the experiment). Other aerosol species (B compounds, Ag, etc.) would be determined later, pending small-scale test results. Steady-state, thermal-hydraulic conditions would prevail, with a continuous standing H₂ flame in the large 850-m³ CSTF vessel. The twofold purpose of the test is (a) to study iodine retention in the various parts of the piping and simulated RCB and (b) to determine the conversion of CsI to I₂ and other volatile species in the H₂ flame. Note that only soluble aerosol species are used so that aerosol deposition (not including chem-sorption) in the pipes between the aerosol generator and containment is expected to be small.

The second test will be an extension of test LA-1 (see document LACE TP-4) with direct injection of I₂ and CH₃I into the containment volume. Following an intact containment for a period of 1-2 hours, a vent would be opened to the environment. The test will examine iodine retention, partition, and volatilization phenomena. A summary of the proposed tests is given in table 5.
Table 5

PROPOSED PHASE B IODINE EXPERIMENTS

<table>
<thead>
<tr>
<th>Test</th>
<th>Analogue</th>
<th>Aerosol</th>
<th>Conditions</th>
<th>Purpose</th>
</tr>
</thead>
<tbody>
<tr>
<td>B1</td>
<td>LA-1</td>
<td>CsI</td>
<td>Steady-state</td>
<td>Determine retention of CsI under basic conditions</td>
</tr>
<tr>
<td>B2</td>
<td>LA-2</td>
<td>CsI</td>
<td>Steady-state with standing H\textsubscript{2} flame**</td>
<td>Determine the retention and conversion of iodine to volatile species</td>
</tr>
<tr>
<td>B3</td>
<td>LA-1</td>
<td>HI</td>
<td>Hold at 115°C, 2 bar for 1-2 hours, then vent through filter</td>
<td>Determine retention of iodine species under acidic conditions</td>
</tr>
</tbody>
</table>

*Alternative is to introduce low levels of H\textsubscript{2} during the test period, burn it and detect changes in fission product and aerosol behavior during and after burning.

It is perceived at this time that the required experimental conditions for tests B1 and B2 will vary as more information is learned from small-scale tests and theoretical studies as part of the support programs during the early phase of the ACE program. Therefore, the proposed test matrix should be expected to be somewhat fluid in nature; that is, the Technical Advisory Committee will be expected to present a revised plan, if necessary, to the Project Board for approval prior to the actual performance of these tests.

Special Instrumentation. The proposed tests are expected to be difficult, particularly in regards to the detection and analysis of iodine species. In addition to the standard instrumentation available for the other tests, the project proposes to use the following devices:

a. Maypact holders with
   --2 glass fiber filters
   --2 Ag membranes
   --1 charcoal paper
   --2 charcoal granule stages (non-KI impregnated)
   (See CSE experiments documentation for further details.)

b. Total Reflectance X-ray Fluorescence (TXRF) technique. (See F. Hegedius et al. paper entitled "Low-Level Iodine Detection by TXRF," 35th Annual X-Ray Conference, August 4-8, 1986, Denver.) The TXRF method does not involve radioactive tracers. It has a very high sensitivity for iodine and involves relatively easy and straightforward sample preparation.
Phase C - Large Scale Reactor Material and Fission Product Release MCCI Experiments

The objectives of the work in phase C are as follows:

A. Measure the release of less-volatile (refractory) fission products, e.g., lanthanum, barium, strontium, and other compounds during the molten corium concrete interaction (MCCI) that may occur during postulated severe accidents.

B. Measure the physical and chemical character of the aerosols generated from the MCCI.

C. Measure the heat transfer in all directions from the molten corium, i.e., to the side, bottom (concrete), and to the atmosphere; measure the concrete ablation rate.

D. Analyze the data obtained in the tests to enable comparison to various analytical predictions.

Background

Postulated risk-dominant accidents, by current definition, lead to a core meltdown. An accident starts with postulated faults, which result in undercooling of the core. This in turn leads to core heatup, degradation, and eventual melting. This in-vessel progression of the accident ends with the failure of the vessel by the attack of the molten core on the lower head. The molten material consisting of UO$_2$ fuel, Zircaloy clad and stainless steel from the vessel internals and their oxides is ejected out of the lower head breach into the containment cavity, and the ex-vessel phase of the accident ensues.

Fission products contained in the UO$_2$ fuel are subject to release while the core heats up during the in-vessel progression of the accident. Indeed, it is very likely that most of the volatile fission products, i.e., iodine (I), cesium (Cs), and tellurium (Te), would be released during the core heatup phase. Tests at Oak Ridge National Laboratory have indicated that some fraction of the tellurium inventory may be retained in the melt if some of the zirconium escapes oxidation during the core heatup and meltdown periods. This retained Te can be released during the ex-vessel progression of the accident.

The initial ex-vessel events are largely determined by two sequence-dependent parameters, namely: 1) the vessel-to-containment pressure difference which determines the vigor of both the corium ejection as well as the ensuing vessel blowdown, and 2) the possible presence of a sufficient water mass in the cavity region to cause even a partial quench of the molten core material. A vessel-to-containment pressure difference of a couple MPA or greater will likely disperse much of the corium from the cavity to other parts of the containment due to entrainment and sweepout with the steam/hydrogen mixture from vessel blowdown. Assuming the containment withstands any possible liner ablation or atmosphere heatup accompanying the dispersal, the debris would likely spread out over large surface areas with relatively small layer thicknesses such that its temperature would be too low for any significant interaction with the concrete.
In the MCCI studies, attention is focused on those sequences in which the corium largely remains in the reactor cavity or pedestal region following failure of the vessel head. In this case, the corium layer is of sufficient depth that high corium temperatures can be sustained from decay heat generation (plus exothermic oxidation). The molten corium, if uncooled, attacks the concrete basemat. The molten corium concrete interaction results in decomposition and melting of the concrete with production of large quantities of carbon dioxide and steam. A cavity forms, advancing both axially downwards and radially outwards in time. Gas bubbles and gas streams from the concrete will pass through the melt layer and emerge at the top.

The carbon dioxide and steam react with the contents of the corium melt to produce a variety of chemical reactions and resulting products. For example, they progressively oxidize the metals in the melt to produce heat, which augments the decay heat being generated. The gases, fission products, and other materials in the melt form various chemical compounds which may vaporize and be carried away with the flowing gases. These vaporized materials after emerging from the corium melt will form an aerosol source as they condense in the containment atmosphere.

Interaction of corium with concrete takes place in four stages, namely:

Stage 1 - Corium Jet Impingement

The corium flows from the bottom of the failed reactor pressure vessel (RPV) and impinges on the concrete base, splashing and spreading out on the base. The basemat is rapidly eroded by the impingement heat flux, but the duration of the jet impingement is relatively short. An aerosol "spike" occurs at this time. The corium temperature may be reduced by partial quenching if water is present in the cavity. The corium temperature will be additionally reduced due to the impingement heat flux into the concrete and the heat sink provided by the concrete decomposition products mixing with the corium. Hence, the initial conditions of the MCCI are influenced by heat losses and dilution of the melt which occurs during the impingement stage.

Stage 2 - Early Aggressive Interaction Stage

Oxidation of metallic species in the corium, notably zirconium, is a major contributor to the internal heat generation during this stage. The corium is spread to a nominally uniform depth on the available floor area in the cavity. The lateral dimension of the cavity is typically much larger than the depth of the corium layer, and hence the MCCI is well characterized as a one-dimensional process. The chemical energy plus decay heat are lost from the melt layer by heat transfer processes which typically involve a downward flux into the basemat concrete, an upward loss by radiation and convection to the overlying atmosphere and structure, and an upward loss by transport of the gases flowing through the melt layer. The early aggressive interaction stage is characterized by relatively high melt temperature, relatively rapid downward erosion into the concrete, high gas flow through the melt layer from decomposition of the concrete, boilup of the melt layer and possibly good mixing of the melt constituents due to the high gas flow, and releases of aerosol and fission products.
Stage 3 - Long-term Erosion Stage

Oxidation of metallic species will eventually become complete, and internal heat generation will occur by decay heating only. The fully oxide melt has been considerably diluted by concrete decomposition products by this time. The corium temperature is much lower, but the corium solidus temperature has also been reduced by addition of the concrete decomposition products. The concrete erosion continues at a reduced rate, and the slower gas release produces a bubbly flow regime. The aerosol and fission product release is small, but continues for a long period of time.

Stage 4 - Solidification Stage

The temperature of the melt eventually reduces to the corium solidus temperature and thick crust zones are formed. Downward penetration occurs at an exceedingly slow rate. Aerosol generation becomes negligible.

Source Term Issue

The magnitude, content, and the physical and chemical character of the MCCI aerosol source are extremely important in estimating the source term due to postulated accidents. In addition to these parameters, the timing of the MCCI aerosol release, relative to that of the fission product aerosol release from the primary system, and to that of the containment failure, is of great importance. If the timing is opportune, the relatively copious MCCI aerosol will help remove the relatively dilute aerosol that may be discharged from the vessel on revolatilization of the fission products deposited on the primary system surfaces. Similarly, if containment failure does not follow soon after the start of the MCCI, there is little danger of having a large airborne source in the containment at the time of containment failure, since natural processes of aerosol removal from the containment atmosphere would have been active for a long time.

A comparison of the risks calculated recently by the IDCOR study with those calculated by the NUREG-1150 is shown in Fig. 4 for the Peach Bottom Mark I BWR Station Blackout Scenario. It becomes clear immediately that the fission products Te, Sr, Ba, La, and Ce (released during the MCCI) dominate the risk estimates and account for the differences in the risks calculated by the IDCOR and the NUREG-1150 studies.

MCCI Model and Code Development

Corium concrete interactions are modeled by several codes developed somewhat independently. The German codes WECHSEL and KAVERN and the U.S. code CORCON are similar in concept but have differences in submodels. The code MAAPX developed under Industry Degraded Core (IDCOR) program employs somewhat different physical concepts about the MCCI. Predictions from these codes for the thermal-hydraulic parameters in typical MCCI events during postulated accidents have been very different from each other. Indeed different heat transfer models employed in CORCON code have led to much different predictions of the temperature histories of the corium melt.

The CORCON code models the MCCI chemistry and thermal-hydraulics in a coupled fashion. However, the chemical treatment is limited in its scope and extent, and a model named VANESA has been developed at Sandia National Laboratories.
in which a chemical-kinetic treatment involving many chemical reactions has been employed. The VANESA code has been used for prediction of the MCCI aerosol source for some postulated accident scenarios. Alternatively the code SOLGASMIX has been employed to estimate the MCCI source term and to perform sensitivity studies. The predictions obtained from these studies have been different than those from VANESA code.

MCCI Experiment Programs

In the past few years a number of experiment programs have begun in which actual molten core materials are used in MCCI tests. These experiments are integral in nature, i.e., the entire physical process is observed and the overall concrete erosion pattern and gas generation rates are determined. Previously, some large-scale (50-500 kg melts at 1800-2500K) tests were performed at Sandia, however, these MCCI experiments were mainly scoping in nature, with limited detailed measurements taken, particularly in regard to aerosol release. The recent BETA experiments in Germany by Alsmeyer, et al., have investigated the MCCI under steady-state conditions in order to get better data on the melt temperature history, erosion rate, gas production, and aerosol release.

The German BETA experiments and the earlier Sandia experiments employed mixtures of stainless steel and aluminum oxide to represent the metallic and the oxide phases of corium and did not employ any fission products. Thus, the chemistry of the MCCI was not represented. The current predictions of the consequences of the MCCI vary so much that it is clear that the thermal hydraulics and the chemistry of the MCCI is not well understood. Further experimentation utilizing real reactor materials is needed, and, in particular, the validation of the methods is only just starting with respect to the fission product source term from corium concrete interactions.

EPRI/ANL MCCI Program

Work is presently under way on small-scale MCCI experiments involving a 20 cm x 20 cm cross section of corium and 25-30 kg corium mass. The experiments are addressing the thermal-hydraulics, chemistry, and aerosol/fission product releases during the second and third stages of the interaction. In the long-term erosion stage, oxidation of metallic species of the corium is essentially complete and internal heat generation is by decay heating only. This is modeled by an experimental technique in which direct electrical heating (DEH) is used for internal heat generation. The oxide layer consists of UO$_2$, ZrO$_2$, Fe$_2$O$_3$, Cr$_2$O$_3$, NiO, CaO, SiO$_2$, and nonradioactive mockups of the refractory oxide fission products La$_2$O$_3$, BaO, SrO, CaO$_2$ plus Te species. The melt layer is in contact with an instrumented concrete slab; presently slabs of limestone/common sand concrete (Zion Station composition) are being used. Decomposition of the concrete not only introduces CaO and SiO$_2$ into the oxide layer, but also sparges the melt with H$_2$O and CO$_2$ decomposition gases. A series of developmental experiments has been completed which have been successful in sustaining an internally heated melt layer atop concrete causing concrete decomposition, erosion, and gas release. There have been steady improvements in the techniques for conducting these tests and in the instrumentation used for melt layer temperature measurement and gas and aerosol diagnostics.
The early aggressive interaction stage involves the oxidation of metallic constituents initially present in the corium as the melt is sparged by the concrete decomposition gases. The oxidation heat source is a major contributor to the internal heat generation during this interaction stage. An intermediate-scale test is under way to examine the possible enhanced fission product release during this stage owing to reduction of LaO, BaO, SrO and CeO\textsubscript{2} by H\textsubscript{2} and CO in the sparge gas. (Metal oxidation causes reduction of the H\textsubscript{2}O and CO\textsubscript{2} gases to H\textsubscript{2} and CO, respectively.) This is being mocked up in the experiment by changing the initial corium composition, deleting CaO and SiO\textsubscript{2} and adding metallic zirconium and stainless steel.

**Large-Scale MCCI Program**

The experiment approach presently employed for the small-scale (30 kg corium) tests is being adopted for a new facility to perform large-scale MCCI tests. This facility will accommodate tests of 50 x 50 cm cross section, amounting to - 300 kg corium mass. This scale-up is needed to satisfy both technical and acceptance criteria for MCCI investigations, namely:

- to satisfy the criterion that the lateral dimension of the interaction zone be sufficiently large to assure a mechanically unstable upper crust,
- to satisfy the criterion for maximum one-dimensional (upward/downward) heat and mass transport (although the option to perform 2-D tests is preserved), and
- to satisfy the criterion that the melt layer be of sufficient depth for realistic chemical processes to occur. Although the lateral dimensions of the experiment are only 50 cm, it is believed that the proposed experiments will provide an excellent full-scale test of both the thermal hydraulic and the chemical phenomena occurring in MCCI.

A new experiment facility is under construction at Argonne National Laboratory for performing the large-scale MCCI tests. This facility is located in an area already having the necessary support capabilities such as overhead crane, scrubber for final cleanup of exhaust gases, controlled access, radiation protection coverage, and facilities for storing and handling radioactive materials (in this case UO\textsubscript{2}). A leaktight cell has been constructed to house the experiment apparatus and assembly/disassembly areas (Fig. 5). The cell is of welded I-beam framing enclosed with welded sheet-metal plates. It has a removable top cover and front passageway for equipment entry. Junction boxes and feedthru panels are provided for instrumentation, control, and electrical lines; feedthru's and piping are provided for the apparatus cooling water, various gas systems, and apparatus exhaust stack. Viewports are provided in the wall for visual observation and video recording during tests. New building power feeds and electrical service have been provided. A 300 KVA transformer has been acquired for DEH internal heat generation in the corium and has been installed and connected to the electrical service and to the experiment apparatus. A digital data acquisition system controlled by an IBM PC/AT and an operating console have been installed together with interconnecting wiring. The exterior of the facility is shown in the photo in Fig. 6.
The principal experiment equipment to be used for the large-scale MCCI tests has also been constructed. This consists of a secondary leaktight liner located inside the cell, the experiment apparatus, and the forms to cast the instrumented concrete blocks. The secondary liner has been installed inside the cell, and feedthru's and junction boxes have been completed (Fig. 7). The experiment apparatus has also been completed (Fig. 7), including the interaction vessel which will eventually contain the concrete block and the corium mass during an experiment. The walls of the interaction vessel are cooled to maintain a constant temperature outer boundary; the cooling water is instrumented to derive the directional heat loss. Two sides of the apparatus are actually tungsten electrodes which provide the current flow for internal heat generation. There is an instrumented, funnel-shaped top cover to duct the gases and aerosol into the exhaust stack. The top cover is instrumented to assess upward heat loss, and has ports for photographing the top surface of the corium layer and for optical pyrometry. The apparatus is leaktight so that gases given off by the interaction process are ducted into the exhaust stack where they will be diluted by a carrier gas for transport to the gas and aerosol collection equipment. The gas and aerosol diagnostics system is illustrated in Fig. 8, and equipment presently used in the smaller scale tests is shown in Fig. 9.

The forms for casting the concrete specimens and for supporting internal instrumentation have been fabricated. The internal instrumentation includes an array of Type K thermocouples to measure the downward rate of concrete erosion plus high-temperature thermocouples in tungsten wells at various depths in the concrete to measure the corium-melt layer temperature as downward migration proceeds.

**PHASE C DESCRIPTION**

**Approach**

The development of the large-scale MCCI apparatus is planned to be completed by the end of 1987. This work includes not only pretest calibrations, checkouts, and debugging, but also a first developmental-type 300 kg experiment. The principal objectives of this first experiment are to check for proper integral operation of the equipment, to assess the adequacy of operational procedures, and to assess the performance of the various diagnostics systems. The facility will be ready for the "production" stage of the experiment program beginning in January, 1988.

The major objective of this work is to provide experimental information that can be used for comparisons with model/code predictions. Test results will be analyzed in such a manner as to convert experiment data into a form which is compatible with the needs of the analysts. There is no intent to develop new models or new codes as part of this program. The objective is to obtain experimental information for assessments of current and improved versions of core-concrete interaction and the MCCI aerosol/fission product release codes.

**Experiment Information**

Posttest evaluation of experiment data consists of basic data processing plus the analysis and examinations required to evaluate the data, where the principal data include the input power; temperatures in the interaction layer,
concrete, and water/gas systems; gas concentrations at various times; and aerosols collected at various times. Posttest physical examinations are planned for each test in order to:

i) characterize the corium/concrete melt layer and ejected materials from several locations for such features as density, porosity, interfacial configuration, evidence for mixing vs. stratification, chemical composition, mass balances, and crust thicknesses;

ii) characterize the molar fractions of gases in the gas samples; and

iii) characterize the aerosols from the various collection devices, by (1) identifying elements and chemical species present with particular emphasis on identifying fission products, and (2) measuring the particle size distributions.

The results of posttest physical examinations plus test data will be used to evaluate MCCI features such as:

i) specific internal heat generation during the test, plus chemical energy release when applicable;

ii) temperature-time history for the molten interaction zone;

iii) downward erosion into the concrete vs. time;

iv) superficial gas velocity from concrete decomposition vs. time;

v) time variation of release rates of steam and the various gases from the MCCI layer;

vi) overall aerosol mass release rate vs. time; and

vii) estimate of aerosol mass release rate of fission products and other selected materials.

Test Matrix

Eight tests are planned to be run during the production stage beginning in 1988. The tests are conducted at approximate three-month intervals. Conditions for the eight tests are given in Table 6. The tests are planned to be nominally one dimensional; i.e., the heat and mass transport processes are upward and downward only. In this approach the walls of the apparatus are not made of concrete, and there is no sideways erosion. This is justified because in the prototypic case, the corium layer depth, even if boiled up by the sparging gas, is considerably less than the characteristic lateral dimension of the cavity or pedestal region. All tests utilize the basic fission products LaO₂, BaO, SrO, and CeO₂ and tellurium compounds in the corium. The parameters varied in the Table 6 test matrix include the type of concrete (limestone/common sand, basaltic, limestone/limestone), the extent of in-vessel oxidation (greater metallic zirconium and stainless steel are present for less in-vessel oxidation), possible presence of control rod material such as B₄C, variation in initial corium temperature, and the possible quench if water is flooded atop the interacting masses. The first test in the series of large-scale tests, designated L-1, is regarded as the reference case.
The reference conditions include the use of limestone/common sand concrete, early aggressive interaction stage characterized by 70% in-vessel oxidation (30% metallic zirconium and stainless steel in the corium), and 2500K initial corium temperature. For this case there are no concrete decomposition products in the initial corium mixture, no control rod material, and no attempt to quench the layer with water. (However, the cooldown behavior following the sudden shutdown of the test operation by interruption of current flow at a selected time will be monitored and will serve as an important point of reference for assessing the quenching effect of the sudden introduction of water in a subsequent test and provide data for later stages of the MCCI.) Subsequent tests L-2 through L-8 feature variations in important MCCI controlling parameters as shown in Table 6.

Relationship of Test Matrix to MCCI Issues

1. **Effect of Type of Concrete.** Addressed by using the three principal types of concrete, i.e., limestone/common sand, basaltic, and limestone/limestone, in the series of tests C-1, C-2, and C-3, respectively. The different concretes have different chemical compositions, different melting points, and release widely differing amounts of gas during decomposition. The specific objective is to examine the effect of the concrete on the interaction thermal hydraulics, chemical interactions, and aerosol/fission product releases during the aggressive interaction stage when the thermal hydraulic and chemical effects on the fission product releases have the greatest potential significance.

2. **Effect of Variation of Initial Corium Temperature.** The initial corium temperature has an important influence on the calculated releases of fission products because of the exponential dependence of vapor pressures of important species with temperature. There is considerable uncertainty in the temperature of the corium flowing from a failed RPV. What is really important is the temperature-time history during the initial aggressive interaction stage, and this is planned to be addressed by the C-1, C-2, C-4, C-5 series of tests using initial corium temperatures of 2500, 2200, and 2000K, respectively.

3. **Effect of Greater Metallic Constituent in Corium.** Zirconium oxidation plays a key role affecting the MCCI temperature and fission product releases during the early aggressive interaction stage. This includes the oxidation heat source which may far exceed the nuclear decay heating during this stage, and it also includes the important reduction of the concrete decomposition gases H_2O and CO_2 to H_2 and CO, respectively. The proposed reduction of the refractory oxide fission products to species of greater volatility because of the presence of H_2 and CO sparge gases is a key issue requiring particular attention. This will be addressed by test C-6 with comparison to reference test C-1; the metallic constituent of the corium will be increased from nominally 30% to 70% to examine the effects on MCCI thermal hydraulics, chemical interactions, and fission product release.
<table>
<thead>
<tr>
<th>Test No.</th>
<th>Interaction Stage (1)</th>
<th>Concrete Type (2)</th>
<th>In-Vessel Oxidation, % (3)</th>
<th>Concrete Decomposition Products, % (4)</th>
<th>Fission Products (5)</th>
<th>$J_{10}$</th>
<th>Initial Melt Temperature, K</th>
<th>Water Quench</th>
</tr>
</thead>
<tbody>
<tr>
<td>C-1</td>
<td>2</td>
<td>L/S</td>
<td>70%</td>
<td>-</td>
<td>Yes</td>
<td>-</td>
<td>2500</td>
<td>-</td>
</tr>
<tr>
<td>C-2</td>
<td>2</td>
<td>B</td>
<td>70%</td>
<td>-</td>
<td>Yes</td>
<td>-</td>
<td>2500</td>
<td>-</td>
</tr>
<tr>
<td>C-3</td>
<td>2</td>
<td>L/L</td>
<td>70%</td>
<td>-</td>
<td>Yes</td>
<td>-</td>
<td>2500</td>
<td>-</td>
</tr>
<tr>
<td>C-4</td>
<td>2</td>
<td>B</td>
<td>70%</td>
<td>-</td>
<td>Yes</td>
<td>-</td>
<td>2200</td>
<td>-</td>
</tr>
<tr>
<td>C-5</td>
<td>2</td>
<td>L/S</td>
<td>70%</td>
<td>-</td>
<td>Yes</td>
<td>-</td>
<td>2000</td>
<td>-</td>
</tr>
<tr>
<td>C-6</td>
<td>2</td>
<td>B</td>
<td>30%</td>
<td>-</td>
<td>Yes</td>
<td>-</td>
<td>2500</td>
<td>-</td>
</tr>
<tr>
<td>C-7</td>
<td>2</td>
<td>L/S</td>
<td>70%</td>
<td>-</td>
<td>Yes</td>
<td>Yes</td>
<td>2500</td>
<td>-</td>
</tr>
<tr>
<td>C-8</td>
<td>2</td>
<td>L/S</td>
<td>70%</td>
<td>-</td>
<td>Yes</td>
<td>-</td>
<td>2500</td>
<td>Yes</td>
</tr>
</tbody>
</table>

Notes:  
(1) Interaction stages: 1 = jet impingement, 2 = early aggressive interaction (prior to completion of metal oxidation), 3 = long term erosion (following metal oxidation), 4 = hot solid/concrete interaction.  
(2) Concrete type: L/S = limestone/common sand, B = basaltic, L/L = limestone/limestone.  
(3) Pertains only to tests examining early aggressive interaction stage; 70% in-vessel oxidation indicates 30% Zr, SSs are in metallic form in corium mixture; similarly, 30% in-vessel oxidation indicates 70% Zr, SSs are in metallic form in corium mixture.  
(4) Refers to percentage of melt layer comprised of oxides of concrete decomposition origin, particularly SiO$_2$ and CaO; only pertains to tests examining long term erosion stage.  
(5) Includes La$_2$O$_3$, BaO, SrO, CeO$_2$, Te compounds.
4. **Effect of B₄C in Corium.** The presence of B₄C in the melt is thought to have a similar effect as that of greater metallic zirconium content. The B₄C interacts chemically with steam to form H₂ and CO. Hence its presence may be predicted to chemically reduce fission product refractory oxides to species of greater vapor pressure. This will be addressed by test C-7 with comparison to reference test C-1.

5. **Effect of Water Flooded Atop Corium Layer.** The capability of a water layer flooded atop the corium layer to mitigate the melt-concrete interaction and the concrete erosion is uncertain, although there would be the beneficial effect of fission product scrubbing. The possible transient quenching of the corium by addition of water at the top will be addressed by test C-8 with comparison to reference test C-1. Pretest analysis will be required to determine the feasibility of DEH as a steady-state heat source with the presence of water. If this does not appear feasible, the approach will be to establish conditions similar to test C-6 such that chemical energy release is the predominant mode of internal heat generation; DEH would be used to initiate the operation until the concrete ablation is such that Zr oxidation is the predominant heat source. At that time the water can be flooded into the system and its effect on the DEH component of heating will not play a major role in the overall energy balance. Special pretest analysis will be required for this test, and there will need to be special provisions to handle the large steam efflux.

**SUPPORT PROGRAM**

The ACE program, like the LACE program, will require a significant amount of cosponsor support to ensure its success. Each cosponsor organization will be asked to roughly-match its direct project contribution with a support program that it undertakes in its home country laboratories. Areas where such work will be needed include the following:

1. **Chemistry**
   --Test planning
   --Sample analysis
   --Theoretical evaluation

2. **Instrumentation**
   --Particle detectors (photometers, spectrometers, imaging systems, etc.)
   --Thermohydraulic instruments (calorimeters, thermocouples, etc.)

3. **Model Development and Code Comparison**
   --Theoretical studies on fission product behavior
   --Analysis and validation of existing codes/models
   --Test planning
4. Thermal-Hydraulic Analysis
   --Test planning
   --Analysis of pool and containment conditions
   --Access to atmospheric condensation work

5. Related Filter Data
   --Access to existing data (pool scrubbing, sand beds, etc.)
   --Analysis and interpretation of existing data
   --Models on aerosol behavior in bubbles and in gravel systems
   --Formation of organic iodides in dry sand/gravel systems

In addition, capable on-site representatives to help run the experiments, to analyze data, and to write test reports will be needed at the ANL and HECL facilities.

PRO-FORMA BUDGET AND SCHEDULE

The preliminary budget and schedule for the ACE program is given in table 7. The budget is tentative pending its formal approval by the Project Board and is subject to modification depending on the number of participants and their level of contribution. Details of the budget categories will be developed in conjunction with recommendations of the Technical Advisory Committee concerning the precise technical definition of the experimental test conditions and submitted on a test-by-test basis to the Project Board for concurrence. The schedule is also subject to modification as the tests proceed. It should be noted that phases A/B and phase C require different equipment and facility access. It is anticipated, therefore, that phase A could be started in parallel with phase C. Other work, as has been already noted, may be desirable extensions to the main program, but would be contingent on obtaining a high level of cosponsorship in the ACE program.
### Table 7
ACE PROGRAM BUDGET AND SCHEDULE

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<th>Phase</th>
<th>Description</th>
<th>Date*</th>
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<td>Start of experiments</td>
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<td>2.</td>
<td>Complete initial test matrix (A1-A12)</td>
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<td>Complete continuation program (A13-A18)</td>
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<td>4.</td>
<td>Complete draft final reports</td>
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<td><strong>Phase B</strong></td>
<td>Iodine Retention Experiments</td>
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<td>1.</td>
<td>Start of planning and instrumentation</td>
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<td>2.</td>
<td>Start of main experimental period</td>
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<td>3.</td>
<td>Test B-1</td>
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<td>4.</td>
<td>Test B-2</td>
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<td>5.</td>
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<td><strong>Phase C</strong></td>
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<td>1.</td>
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<td>2.</td>
<td>Complete and analyze tests C1-C3</td>
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<td>Complete and analyze tests C4-C7</td>
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<td>Complete test C8**</td>
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*Number of months after official start of program. Current goal is to begin in fall, 1987.

**Test C8 is not part of the current budget and would require (1) better progress in phase C than is currently assumed, (2) use of contingency funds, or (3) additional resources not reflected in the current budget.
Figure 1. View of the LACR Facility showing the Venturi and Fibrous Filters.
Figure 2. View and operating details of the Submerged Gravel Bed Scrubber at the LACE Facility.
Figure 3. Large SGS Scrubber at the Large Sodium Fire Test Facility.
Figure 4. Comparison of IDCOR vs. NUREG-1150 Risk Calculations for Peach Bottom Station Blackout Sequence.
Figure 5. Experiment Facility for EPRI/AML Large-Scale MCC1 Tests.
Figure 6. Photo of Large-Scale MCCI Facility.
Figure 7. Experiment Apparatus for Large-Scale MCCI Tests.
Figure 8. Gas and Aerosol Diagnostics System for Large-Scale MCCi Tests.
Figure 9. Photograph of Equipment Used in the Gas and Aerosol Diagnostics System of Small-Scale MCCI Tests.
References


18. A. Butland, M. Mignanelli, P. Potter and P. Smith, "The Vaporization of Chemical Species and the Production of Aerosols During a Core Debris/Concrete Interactions," Proceedings of the CSNI Specialist Meeting on Core Debris-Concrete Interactions, EPRI NP-5054-SR, February 1987.


Another proposed international project closely related to those of the NEA in the source term and containment performance area is the Advanced Containment Experiments (ACE) Programme. This will be an extension of the already completed LWR Aerosol Containment Experiments (LACE) Programme in which the United States, through the Electrical Power Research Institute co-operated with organisations in a number of other OECD-NEA Member countries. The NEA is not at this stage formally involved with the ACE Programme but there has been close mutual interest between LACE and NEA programmes. EPRI which has taken the lead in putting together the ACE programme proposal, has requested that we appraise you of its objectives.

Three research areas have been identified for further work.

1. **Containment Filtration**: Parametric studies are proposed to develop a data base for comparing various concepts for filters to remove fission products during accident depressurisation situations.

2. **Integral Iodine and Fission Product Retention Tests**: Integral tests are proposed to demonstrate retention of iodine and fission products in the primary system, containment system and during simulated release to the atmosphere. Several iodine species and important reaction partners such as boron and silver will be involved to get data not obtained in the earlier Harviken, DEMONA or LACE programmes.

3. **Large Scale Molten Core - Concrete Interaction Experiments**: A series of large scale reactor materials tests and related analysis are proposed to investigate the thermal-hydraulic, chemical and aerosol release processes of molten-core concrete interactions (MCCI). These tests are justified by the large contribution that MCCI is calculated to make to the magnitude of and uncertainty in the overall source term from postulated severe accidents.

**Background information**

Experiments 1 and 2 will be carried out by Battelle Pacific Northwest in the Containment Systems Test Facility at the Hanford Engineering Development Laboratories. This facility was previously used for the LACE Programme. Experiment 3 will be carried out by the Argonne National Laboratory in a new experimental facility.

Countries considering participation include: United States (EPRI, DOE, NRC), Japan, United Kingdom, France, Federal Republic of Germany, Italy, Finland, Switzerland, Spain, the Netherlands and the CEC. (Canada is also a possibility and there is likely to be some direct participation by an industrial group).
ANNEX III

List of TC Participants
NOTIFICATION OF AN AGENCY SPONSORED MEETING

Title of the Meeting: Technical Committee Meeting on Thermal Reactor Safety Research

Dates, inclusive: 9-12 June 1987

Place: Vienna International Centre, Meeting Room IV (Building C, 7th Floor) Ext: 1341

Participants and Designated Member States and Organizations

<table>
<thead>
<tr>
<th>CANADA</th>
<th>ADDRESS ABROAD</th>
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<tr>
<td>Mr. J. Sainsbury</td>
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<td>Mr. M. Hrehor</td>
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<td><strong>FINLAND</strong></td>
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<tr>
<td>Mr. L. Mattila</td>
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<tr>
<td></td>
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<tr>
<td>Mr. R. Zammit</td>
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<td>Mr. H. Banaschik</td>
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<tr>
<td>Mr. L. Szabados</td>
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<tr>
<td>Mr. L. Voross</td>
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| Mr. A. Annunziato                                          | ENEA-TERM/MEP/SISREF  
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km 1,300, 00100 Roma | Etap Hotel Belvedere  
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| Mr. J. Sánchez Gutiérrez                                   | Comisión Nacional de Seguridad  
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Delegación Álvaro Obregón  
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1020 Vienna  
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| **NETHERLANDS**                                            |         |                  |        |
| Mr. K.J. Brinkmann                                         | Netherlands Energy Research Foundation  
P.O. Box 1  
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Tel. 02246-4474  
Telefax 02246-4480 | Hotel Nordbahn  
Praterstrasse 72  
1020 Vienna  
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| **POLAND**                                                 |         |                  |        |
| Mr. E. Józefowicz                                          | Central Laboratory for Radiological Protection  
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Konwaliowa 7  
03-194 Warsaw  
Tel. 11.00 11 ext. 80 | Hotel zu den drei Kronen  
Schleifmühlgasse 25  
1040 Vienna  
Tel. 57 32 89 | 9-12 June 1987 |
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<td>Mr. M. Montes</td>
<td>Medio Ambientales y Energéticas (CINMAT) Avenida Complutense 22 Madrid 28040</td>
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<td>Mr. D. Rafael San Martin</td>
<td>Jefe del Servicio de Centrales Nucleares Ministerio de Industria y Energía Paseo de la Castellana, 160 Madrid 28046 Tel. Ministerio 458 80 10 x 1458 direct 250 69 15</td>
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<td>Mr. D. Haschke</td>
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<td>Mr. J.L.M. Cortez</td>
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